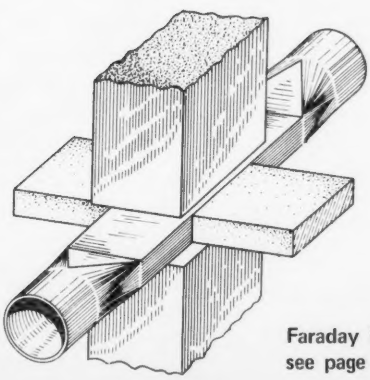
See the note on page ii

REACTOR
AND FUEL-PROCESSING
TECHNOLOGY

A Quarterly Technical Progress Review

Prepared for DIVISION OF TECHNICAL INFORMATION, U. S. ATOMIC ENERGY COMMISSION



Faraday Pump — see page 267

Summer 1969

VOLUME 12

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TECHNICAL PROGRESS REVIEWS

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REACTOR
AND FUEL-PROCESSING
TECHNOLOGY

Vol. 12, No. 3 Summer 1969

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The procedure for the dismantling of the BONUS reactor is presented on page 293.

Superintendent of Documents, U. S. Government Printing Office, Washington, D. C. 20402, $2.50 per year.

Note to Our Readers

In reply to many inquiries, we are pleased to announce that because of widespread interest and support the publication of this journal will continue.

Beginning with this issue we are adding Current Awareness Review articles prepared by the Reactor Technology Section, Division of Technical Information Extension, Oak Ridge.

In addition, beginning soon, we plan to include a major new section made up of detailed Critical Review articles prepared by experts under the aegis of the American Nuclear Society.

Because of the increasing interest in the development of breeder reactors, increased emphasis will be put on this subject area in the future.

We welcome your comments on the journal.

Edward J. Brunenkant

Edward J. Brunenkant, Director
Division of Technical Information
U. S. Atomic Energy Commission

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Current Practices in Refueling Light- and Heavy-Water Power Reactors

By William H. Steigelmann*

In any nuclear reactor, whether it is a low-temperature type used for research and training or the commercial type used for generation of electric power, the fuel-handling systems must provide a safe and effective means of moving, handling, and storing the nuclear fuel from the time it reaches the reactor site until it (or the fuel residue) is shipped off-site for reprocessing or disposal after postirradiation cooling. The importance of these systems is naturally greater from both a safety and an economic point of view in the case of a large commercial nuclear power plant. Not only are the quantities of fuel to be handled quite large (approximately 30 to 50 metric tons per year after the initial fueling) but also the consequences of a malfunction or accident can be much more severe.

With regard to economic considerations, the refueling and scheduled maintenance of a large nuclear power plant requires careful planning and skillful implementation to minimize the time that the unit is out of service. The cost of downtime to the electric-utility organization that owns the plant is typically \$50,000 to \$80,000 for an 800- to 1100-Mw(e) unit. Although on-load refueling has been the general practice for European and Canadian reactors fueled with natural uranium, this choice has been dictated principally by consideration of the excessive "downtime" that would be required with off-load refueling because of the relatively low burnup that occurs with the natural-uranium system. Pressurized- and boiling-water reactors (PWR and BWR), which are fueled with slightly enriched uranium and offered commercially by reactor vendors in the United States and abroad, feature off-load refueling that is scheduled to coincide with the annual outage for plant maintenance.

The foremost requirements in the design of power-reactor refueling systems and equipment are that (1) refueling be accomplished with maximum safety, (2) there be minimum probability of inadvertently damaging the fuel or reactor components, and (3) reactor downtime be minimized. Specifically, the following features are generally believed to be necessary:

- Positive grappling for all fuel- and component-handling devices.
- Travel and load limits to ensure that safe design conditions are not exceeded.
- Procedural and equipment backup provided for use in the event of a malfunction.
- "Fail-safe" equipment.
- Maximum visibility maintained and readout equipment provided to allow the most intelligent operation of the system.

This article is concerned with the procedures and equipment used for refueling large power reactors of either the light-water-moderated (PWR and BWR) or heavy-water-moderated (heavy-water-cooled or boiling light-water-cooled) type.

Light-Water Reactors

In both BWR and PWR systems, the general procedure is to store fresh fuel in a secured area. The fuel assemblies are normally stored dry—not immersed in water. The storage vault is required to provide a safe geometrical spacing for the fuel, in case of accidental flooding. Immediately after arriving at the plant, the crates containing the fuel assemblies are

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examined for indications of damage during shipment. After the fuel assemblies are uncrated, they are directly examined and placed in the storage vault.

Depending upon the fuel-management scheme being followed, a portion (generally between one-fifth and one-third) of the core is refueled off-load during the annual maintenance outage, and the positions of a number of other fuel assemblies are changed within the core. The purpose of this shuffling operation is to achieve the most desirable flux distribution in the core for economical operation of the reactor.

Because the spent fuel is highly radioactive, it is necessary to provide both radiation shielding and a means of removing the internally generated decay heat. The most economical way to accomplish this is to perform all fuel-transfer operations underwater, with the water providing both the shielding and the cooling. Similarly, the spent fuel is stored on-site for a period of a few months prior to shipping to the reprocessing plant to allow the level of radioactivity to decay to some extent. The length of this postirradiation "cooling" period is determined from an economic analysis, finding the optimum trade-off between the increasing accumulated cost of interest charges on the value of the fuel during the period and the decreasing cost of shipping as the level of radioactivity drops off.

The spent-fuel storage pool and the transfer canal between the reactor and the storage pool are usually in the form of reinforced-concrete wells that are lined with a corrosion-resistant steel. Earlier attempts in some nuclear power plants and research reactors to utilize sprayed or painted plastic coatings proved to be unsatisfactory because of difficulty in effecting decontamination or failure to achieve and assure leaktightness.[1-3] In addition to storing spent fuel during the postirradiation cooling period and holding fresh fuel prior to its being transferred into the reactor, the fuel storage pool is also used for holding used control rods, poison curtains, fuel channels, and other components temporarily or permanently removed from the core. In BWR plants, for example, the fuel assemblies are jacketed in removable fuel channels which serve to (1) contain the main flow of coolant within the fuel bundle and (2) guide the cruciform control blades that are located between fuel assemblies. The pool also provides a working space for stripping and replacing these channels and for removing and inserting assemblies of control elements when these are located within the fuel assembly, as is the case with virtually all of the newer PWR plants. The pool is equipped with purification and cooling systems and with viewing devices for examining spent fuel.

Boiling-Water Reactors

Figure 1 shows the general arrangement of the reactor building in a typical BWR plant. Refueling operations are carried out from above the reactor and the dry well of its pressure-suppression containment system. The dryer–separator and fuel storage pools and laydown space for the shield plugs, dry-well head, and reactor-vessel head are located at the level of the refueling floor, and a traveling-bridge crane is provided above it.

Refueling operations are carried out from a movable refueling platform that travels on a set of rails extending along either side of the storage pool and reactor well. A telescoping grapple hoist and its trolley travel on another set of rails approximately 10 ft above the refueling-platform bed. The platform operator rides on the grapple-hoist trolley and controls the movements of the hoist and platform. One lever controls the horizontal motions of both the platform and the grapple-hoist trolley, and the other controls the vertical motion of the telescoping grapple hoist. A viewing tube extending below the water surface of the refueling pool allows the operator to view the position of the fuel grapple without encountering distortions and distractions due to surface reflections. The primary function of the grapple hoist is to transfer fuel between the pool storage racks and the reactor core during the refueling operation. An auxiliary hoist is mounted on the refueling platform to facilitate movement of the control rods. Two jib hoists are located at the work end of the fuel storage pool to permit an operator to handle fuel within his portion of the work area, thus leaving the refueling platform free for general fuel-handling operations. Interlocks prevent lifting exposed fuel beyond minimum water-shielding limits. The hoists can be moved to other locations where mounting sockets have been provided. A socket placed near the new-fuel storage vault allows the use of a jib crane for transferring new fuel from the storage vault to the storage racks within the pool. Other sockets placed around the periphery of the pool are available, and a crane can be moved to one of these locations should the need arise. In addition to the building crane and jib hoists, several hand devices are also provided. These generally include miscellaneous lines, grapple poles, and a hand-actuated grapple.

The first operation after reactor shutdown is to cool the vessel to ambient conditions. The temperature and pressure of the primary coolant are reduced by circulating the coolant through auxiliary heat exchangers. The vessel is filled with water when the

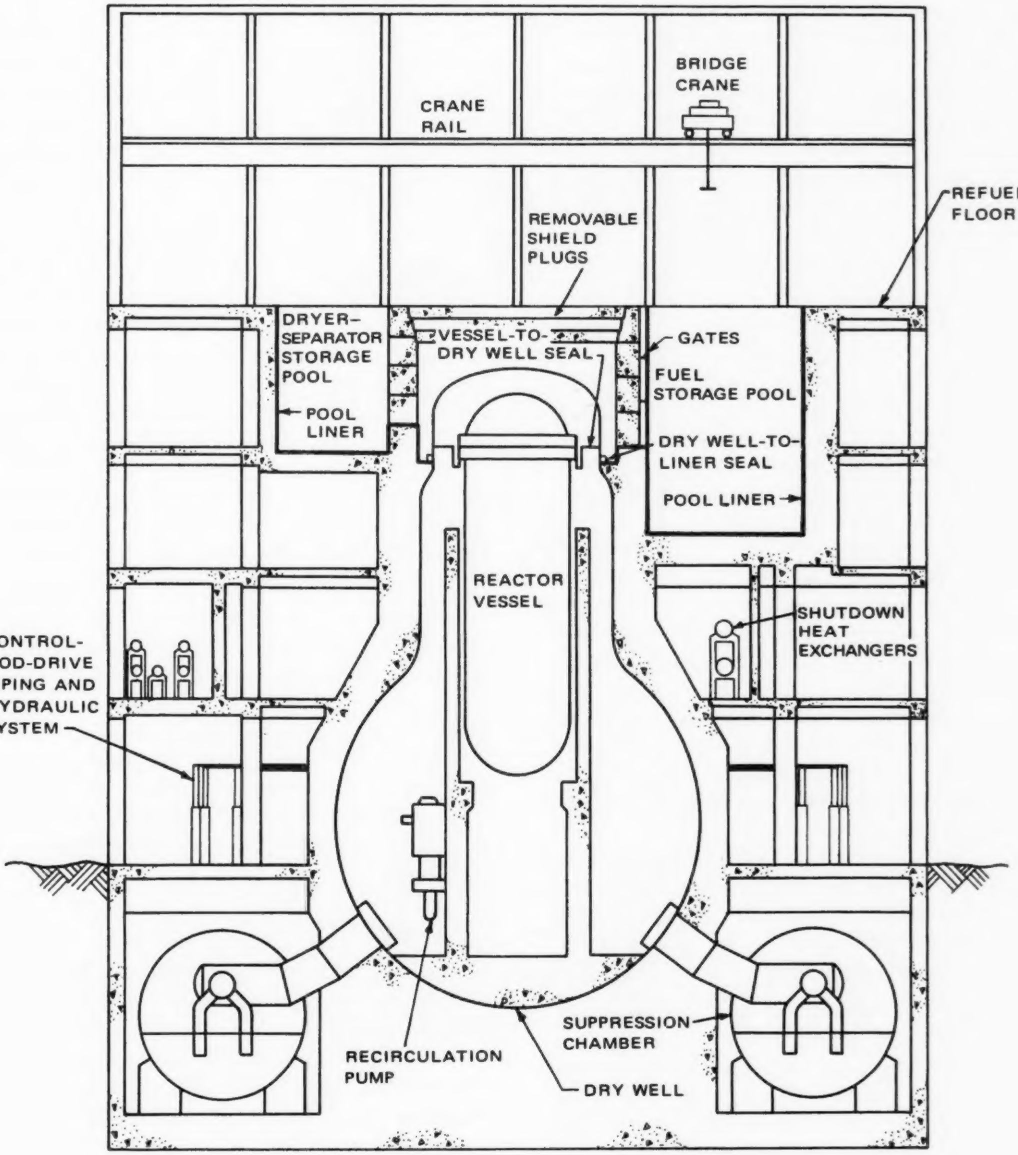


Fig. 1 .Typical arrangement of reactor building in a BWR plant.

pressure has been reduced to atmospheric. During the latter stages of these operations, the shielding blocks between the reactor well and the spent-fuel storage pool and between the reactor well and the dryer–separator pool are removed. The dry-well head is then removed and stored on the refueling floor, and the reactor-vessel-head insulation is removed to further facilitate cooling. The instrument and vent connections on the top head are removed next, and the top-head nuts and studs are removed and replaced with guide pins. The seals between vessel and dry well are then installed, making the reactor dry well watertight above the level of the vessel flange. The top head is then removed and stored on the refueling floor. A seal surface protector is installed on the face of the vessel flange to prevent any damage during the refueling operation. The reactor well, dryer–separator storage pool, and connection canals are now filled to the level of the spent-fuel storage pool. The steam-dryer and -separator-plenum assembly is then removed and transferred by the building service crane to the dryer–separator storage pool. The spent-fuel storage-pool gate is opened in readiness for moving fuel.

During reactor operation the concentration of radioactive off-gases emitted from the core is monitored. If there had been a rise in off-gas concentration prior to the shutdown, a core-survey operation would be performed next. The coolant emanating from the fuel assemblies is sampled ("sipped") to locate specifically which assemblies are leaking. To prevent the spread of contaminants, workers store any leaking fuel assemblies in special fuel containers (rather than placing them with the other assemblies in the fuel storage racks) in the pool.

New fuel assemblies are transferred from the storage vault to the pool prior to, or during, the preceding operations. Spare or previously used channels are placed around some of the assemblies. The sequence of fuel-transfer operations is then as follows:

Defueling. The refueling platform operator aligns the fuel grapple hoist over a fuel assembly in the core and actuates the grapple to grasp the top of the assembly. He then raises the fuel assembly out of the core, transfers it through the refueling canal to the spent-fuel storage pool, and lowers it either into the storage rack or, in the case of leaking fuel, into one of the special fuel containers.

Relocating. A number of fuel assemblies are moved to different core locations. This fuel-shuffling operation is accomplished by the fuel grapple hoist in the same manner as in the defueling procedure.

Reloading. New fuel assemblies from the fuel storage pool are loaded into the vacant core positions. To the extent that extra channels are not available prior to the refueling, channels must be removed from the spent-fuel assemblies as they are removed from the core, inspected, and placed on the new-fuel assemblies. Since these operations are sequential with the others, they tend to lengthen the total period required for refueling. Thus, if refueling is the controlling activity during the downtime, the cost of spare channels must be balanced against the cost of extending the maintenance outage.

Following completion of reloading, physics tests are conducted to verify that the measured values of parameters which describe the core configuration and condition agree with the calculated values. The vessel-closing procedure is essentially the reverse of the opening procedure except that, after the storage-pool gate has been closed and the water has been drained from the canal's reactor well and dryer–separator pool, the walls are scrubbed to remove any contaminants that may have been deposited during the refueling operation. Prestartup tests are then performed to ensure that the fluid systems and containment are leaktight and that all safety systems are operational.

The spent-fuel assemblies removed from the reactor during a refueling outage will undergo channel-removal operations, and the spent-fuel bundles will be returned to the storage racks to allow adequate radioactive decay before being placed in the shipping cask for transfer to a reprocessing plant. The channels removed from the spent-fuel bundles will be inspected to determine if they are reusable; if they are not, they will be inserted in shielded containers and shipped off-site for disposal.

Pressurized-Water Reactors

In many respects the refueling of PWR units is quite similar to the operations followed in BWR plants. The principal difference is that in the PWR it is common practice to locate the spent-fuel storage pool in a building adjacent to the containment structure (which is, in effect, the reactor building). A typical arrangement of the components and equipment in the refueling system is shown in Fig. 2. The principal components in the refueling system are the refueling machine, the transfer system (carriage, tube, and tilt machines), the spent-fuel-handling machine, and a machine or manual equipment to remove or replace the neutron-absorbing control elements. In some plants these elements are cruciform in cross section and are

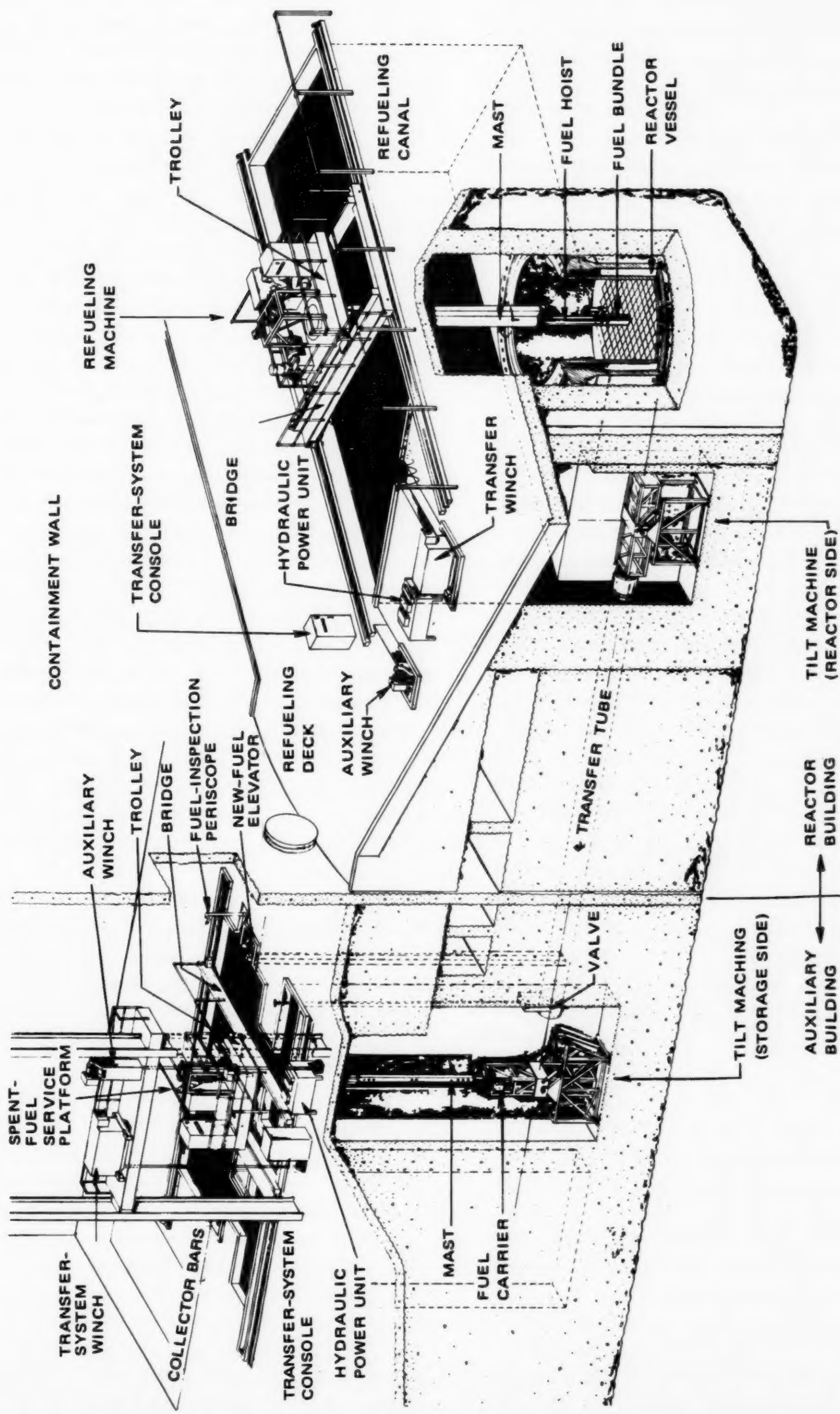


Fig. 2 General arrangement of refueling system in a PWR plant. (Courtesy of Combustion Engineering, Inc.)

located between the fuel assemblies. In all of the newest PWR plants, however, the control elements are in the form of a multiplicity of rods which are inserted within some of the fuel assemblies.

The description of the refueling system presented in the following paragraphs applies specifically to the Palisades Nuclear Power Station,[4] but this is typical of the system used in all PWR plants (although the design of specific equipment varies from plant to plant).

The refueling machine is located within the containment structure on the refueling deck, above the reactor vessel. The basic structure of this machine is a traveling bridge which spans the refueling canal and moves on rails. Mounted on this crane is a trolley that travels in a transverse direction. Installed on this trolley are the display boards, the TV monitor, and the control console that allows an operator to perform and control the refueling operations. In addition, the trolley carries the fuel hoist motor and the hoist cable support. Suspended from the trolley and supported by a thrust bearing is the fuel-handling mast assembly that is angularly oriented by the rotation drive mechanism. Within the mast assembly is the fuel-handling hoist assembly, and this is moved in a vertical direction by a cable which is attached to a swivel at the top of the hoist assembly and which runs over the sheave of the hoist cable support to the fuel hoist drive system. The hoist assembly contains a coupling device and an actuator mechanism to accomplish engagement with the fuel assembly to be replaced or moved.

For refueling, the operator moves the bridge and trolley to position the center of the hoist over the fuel assembly to be removed. This is accomplished by monitoring the rectangular coordinate position of the hoist on readout devices at the control console. The fuel hoist motor is actuated to lower the hoist assembly for engagement with the fuel assembly, and the vertical position of the hoist assembly is monitored on another readout device at the control console. Prior to engagement, the operator checks the tool alignment by viewing the TV monitor, which displays a picture of the top of the fuel assembly. If necessary, minor adjustments in the position of the coupling device can be made by movement of the bridge and/or trolley. After the hoist has been correctly positioned, the operator energizes the actuator mechanism and engages the hoist assembly with the fuel assembly.

A spreading device is attached to the lower portion of the hoist to contact adjacent fuel assemblies and provide clearance during fuel removal and insertion. The fuel hoist motor is then started, and the fuel assembly and hoist assembly are lifted into the trolley mast until a signal on the control console indicates that the refueling machine is ready to be moved. Installation of a new-fuel assembly and transfer of an assembly from one core position to another are essentially the reverse of the procedure just described.

The load on the hoist cable is monitored at the control console during withdrawal or insertion of a fuel assembly to ensure that movement of the fuel is not being restricted, and limits are built into the equipment to ensure that the fuel cannot be damaged. Positive locking between the grab and the fuel assembly is provided by the engagement of the actuator arm in axial channels running the length of the hoist assembly, thereby precluding relative rotational movement and accidental uncoupling even with inadvertent initiation of an uncoupling signal to the actuator mechanism. The drives for both the bridge and the trolley provide close control for accurate positioning, and brakes are provided to maintain the desired position. In addition, interlocks are installed so that movement of the refueling machine is not possible when the hoist is withdrawing or inserting a fuel assembly.

In the fuel-transfer system, the transport-transfer carriage travels within a transfer tube and carries fuel assemblies between the refueling canal in the containment building and the fuel storage pool in an adjacent auxiliary building. Eight large wheels support the carriage and allow it to roll on tracks within the transfer tube. Track sections at both ends of the transfer tube are supported from the pool floor and permit the carriage to be properly positioned in the tilt mechanisms.

During reactor operation the transfer tube is closed by an isolation valve outside the containment and a blind flange inside the containment. The tube is supported by a larger diameter pipe, which in turn is sealed to the containment envelope. The two concentric tubes are sealed to each other with a bellows-type expansion joint.

Two tilt machines are provided in the system: one is installed within the refueling canal inside the containment, and the other, in the spent-fuel storage pool. Each consists of a structural-steel support base from which an upending straddle frame is pivoted. Interlocks are provided to ensure the safe operation of this equipment by (1) prohibiting the lowering of a fuel assembly unless the fuel carrier is vertical, (2) preventing inadvertent rotation of the tilting cylinders while a fuel assembly is being lowered, and (3) deactivating the cable drive so that a premature attempt to move the carriage through the transfer tube cannot be initiated. The tilting of the fuel assemblies

from horizontal to vertical is accomplished by two boric acid-actuated hydraulic cylinders, each of which is capable of performing the operation in the event of the failure of the other. As a backup to this motion, as well as to all other motor-driven operations, manual drives are provided to allow completion of any fuel-transfer procedure in the event of motor failure or a loss of electric power.

For long transfer tubes (those approximately 40 to 80 ft in length), two winch mechanisms are provided to transfer the fuel carrier between the refueling canal and the fuel storage pool. A double cable system is provided to ensure continuity of operation in the event of the failure of one cable. Design of the equipment has been based on the loads that would be imposed in the event that all wheels of the transfer carriage freeze in operation, thus assuring that a fuel assembly will not be trapped within the transfer tube. Load-indication devices are present, and a dual horsepower motor is provided to accomplish both normal and overload modes of winch operation.

Where shorter transfer tubes are compatible with containment-building design, the transfer carriage is driven from either the containment building or the auxiliary building with a single winch mechanism. This simplifies the equipment design and provides for remote access to the carriage during any portion of its travel through the transfer tube.

In the auxiliary building, a spent-fuel-handling machine is mounted on a spent-fuel service platform, which is in turn mounted on a traveling bridge that spans the spent-fuel storage pool. The bridge is supported on rails, and thus the fuel-handling machine can remove spent-fuel assemblies from the tilt machine and transport them to any location within the pool. Similarly, it can pick up fresh fuel, carry it, and deposit it in the tilt machine to initiate its transport to the core.

In plants in which the control rods are inserted into the fuel assemblies, a machine is sometimes provided to facilitate the exchange of control elements between fuel assemblies. This device typically consists of a structural frame mounted on the wall of the refueling canal, a translation drive, and a grapple hoist. At the bottom of the hoist section, an alignment device centers the fuel assembly, and a finger guide maintains the pattern of the individual control rods when the control assembly is withdrawn. The grapple load is monitored, and position readout is provided for both hoisting and translation drives. Interlocks are provided to ensure that translation cannot occur during hoisting, and vice versa.

The reactor cooldown and other preparatory steps are generally quite similar to those described above for the BWR. Two essential differences are that (1) there is no dry-well head or steam dryer–separator assembly to be removed and (2), since the control-rod drives are located on the top head of the reactor vessel (instead of on the bottom head as in the BWR), it is necessary to disconnect the drives from the rods within the core and to disconnect the electrical leads and cooling lines from the drives. After reactor shutdown, new-fuel assemblies can be transferred from the fresh-fuel storage area to the refueling canal within the reactor building in either of two ways: (1) carried dry through the equipment hatch or (2) by way of the spent-fuel storage pool using the fuel-transfer carriage and the fuel-transfer tubes. The sequence of fuel-transfer operations is very much the same as for the BWR. Although the PWR fuel assemblies are not encased in channels, many of the fuel assemblies may contain assemblies of control elements. As described above, these must be removed from the fuel assemblies being discharged from the core and also from some of the ones being moved to other locations within the core and inserted into some fresh assemblies and into some of those being shuffled. To the extent that spare assemblies of control elements are not provided for advance inplacement, these operations must occur sequentially with the others.

Duration of the Refueling Period

The total time required to accomplish the refueling of a large unit [~800 to 1100 Mw(e)], from reactor shutdown until completion of prestartup tests, can be expected to range from 35 to 40 days for the first refueling down to the order of 15 to 25 days after the equipment and procedures have been thoroughly "debugged," and the plant personnel have gained experience in performing a number of refuelings. These time periods are to some extent dependent upon the thermal capacity rating of the unit, since the number of fuel assemblies to be handled is directly proportional to the rating. There are other factors, however, that tend to overshadow the influence of the size of the unit, the fuel management scheme being followed, or whether the unit is a BWR or a PWR. (A BWR has a larger number of fuel assemblies in the core, but generally the fuel-management scheme is such that a smaller fraction of the core must be moved.) It should be noted that the time required for inspection and maintenance of equipment in the turbine plant may set the duration of the annual outage, particularly for units having capacities of 600 Mw(e) or less.[1]

Time estimates much shorter than those given above are frequently quoted by the reactor vendors, but they are based on the use of experienced crews and on the assumption that no special problems occur. Such refueling outage times should be viewed by the plant owner as an absolute minimum that probably will not be realized until the third or fourth refueling. Items that can be expected to affect the time period required for refueling either BWR or PWR units are:

- Crew inexperience.
- Refueling-equipment malfunctions
- Improper or insufficient procedures.
- Problems with reactor fuel elements, vessel internals, or control rods, such as unexpected wear or damage.
- Contamination of work areas by radioactivity.
- Radiation-dose limitations affecting availability of personnel.
- Any unanticipated problems that arise, even if minor, since these will require time for communication and decision.
- Extent of fuel-inspection operations.

There are some factors that can have a different effect on the refueling period, depending on whether the unit is a BWR or a PWR. For example, the BWR is refueled by moving fuel underwater from the reactor, through a canal, to the fuel storage pit—all within one building. The PWR plants, on the other hand, require the remote transport of spent fuel from the containment to the fuel storage pool through a submerged transfer tube. This requires manipulating the fuel element from an upright to a horizontal position, transport of a carriage through a tube, and repositioning to a vertical orientation. Mechanical breakdown of this system could obviously seriously impair refueling operations and cause a severe loss of time. Therefore some PWR plant owners have planned to install two independent transfer tubes and mechanisms. With the BWR, on the other hand, seizure of fasteners or breakage of bolts during the removal or installation of the steam-separator assemblies can be very costly in time. Also, inerting of the containment-system atmosphere is required with the pressure-suppression type of containment system that is virtually always used with the BWR, and there is additional time required to purge this system at the start of refueling and to reinert it prior to plant startup.

Many plant owners who have gone through refueling difficulties with early prototype plants have stated that the best way to achieve short refueling times is to (1) have detailed and accurate work schedules and procedures prepared well in advance of the shutdown, (2) inspect and exercise all equipment prior to the shutdown, (3) have accurate predictions of the radiation levels to be encountered, and (4) have an ingenious master mechanic in the refueling crew to solve the unexpected problems.[5] In summary, the only sensible approach to refueling is to have reliable equipment and procedures and a properly trained crew that can respond intelligently to unexpected situations.

Heavy-Water Reactors

As noted in the introductory discussion, this section will treat both pressurized heavy-water-cooled and boiling light-water-cooled systems. Plant designs from Canada and Germany will be used to illustrate the former class of systems, whereas the principal Canadian and United Kingdom (U. K.) designs will be described for the latter class. All descriptions are summarizations of those presented in Ref. 6.

Compared with an enriched-uranium light-water reactor of the same capacity rating, a much larger number of fuel assemblies are handled per year in the natural-uranium-fueled heavy-water-reactor plants with their lower fuel burnup. As noted earlier, on-load refueling is featured in virtually all of these plants to avoid frequent shutdowns for refueling and fuel shuffling. In addition to requiring a high initial investment, these on-load refueling systems are also potential sources of operational difficulty because of the need to achieve close alignment, to break and remake tight seals in the primary system, and to have complex mechanisms that can perform the required intricate motions; all operations are controlled from a remote location. Because of the possible economic consequences that would result if the only refueling machine at the plant should fail to function properly, many plant owners believe that it is better to accept the added cost of providing a spare machine. The systems involved in the storage of new and spent fuel are quite similar to those used in the light-water plants, and therefore these are not discussed again here.

Pressurized Heavy-Water-Cooled Designs

Pickering Nuclear Power Station. The Pickering Station is a multireactor complex of four 500-Mw(e) units. The natural-uranium-fueled reactors were designed by Atomic Energy of Canada Limited (AECL) for Ontario Hydro, the owner–operator electric-utility organization. The heavy-water coolant and fuel are inside of pressure tubes. Refueling is performed by

inserting new-fuel bundles at one end of a channel in the direction of coolant flow (which is in opposite directions in adjacent channels), the spent fuel being thereby pushed out of the other end of the channel. Each of the fuel channels is sealed at each end by a closure plug. Two fueling machines are used simultaneously, one to insert new fuel and one to receive the spent fuel. The bidirectional fueling system is used to enhance the symmetry of the core flux.

The fueling-machine head (Fig. 3) consists of a pressure housing containing a magazine, a snout to clamp onto a channel, and various drives. The 12-tube rotating magazine carries the fuel bundles and other parts required for refueling. At the aft end of the magazine, there is a mechanical ram for closure-plug and shield-plug removal and a hydraulic ram for fuel insertion.

The fueling machines at each end of the reactor are required to engage with any one of the 390 fuel channels and with the new-fuel and spent-fuel ports in the fuel-transfer bays. Each fueling-machine head is suspended from a trolley that is supported by a bridge structure, as shown in Fig. 4. The necessary Y-coordinate motion is provided by up-and-down motion of the bridge along the two vertical columns. The X-coordinate motion is achieved by moving the trolley back and forth along the bridge; and the Z motion, parallel to the channels, is obtained by reciprocating the head through the suspension. Services for hydraulic and electric functions are carried in catenary-loop service systems.

Fuel is expelled from the magazine by the hydraulic ram, which is operated by connection to a high-pressure heavy-water source. A lower pressure applied to the hydraulic cylinder in the receiving mechanism controls the rate of receipt of spent fuel. The ram motions are monitored by rotary potentiometers.

After inspection, workers load new fuel by hand into a trough aligned with a shielded port leading to the fuel-transfer room. If the radiation level in the transfer room is low enough, the shielding plug is removed from the port, and the new fuel is rammed through into a new fuel loader of the rotary-magazine type. New fuel enters the lowest barrel of the magazine and is discharged by ram action from the topmost barrel into the transfer mechanism. The transfer room is maintained at a partial vacuum so that air will flow from the inspection room into the transfer room.

After the transfer mechanism is fully loaded, it is aligned with the transfer tube connecting the transfer room with the fueling-machine service bay. A lock ring and an O ring are used to make a sealed junction between the transfer mechanism and the transfer tube, and the transfer mechanism is pressurized. The fueling machine then transfers spent-fuel bundles to the transfer mechanism, and, with appropriate indexing, the new-fuel bundles are ejected from the transfer mechanism into the fueling machine.

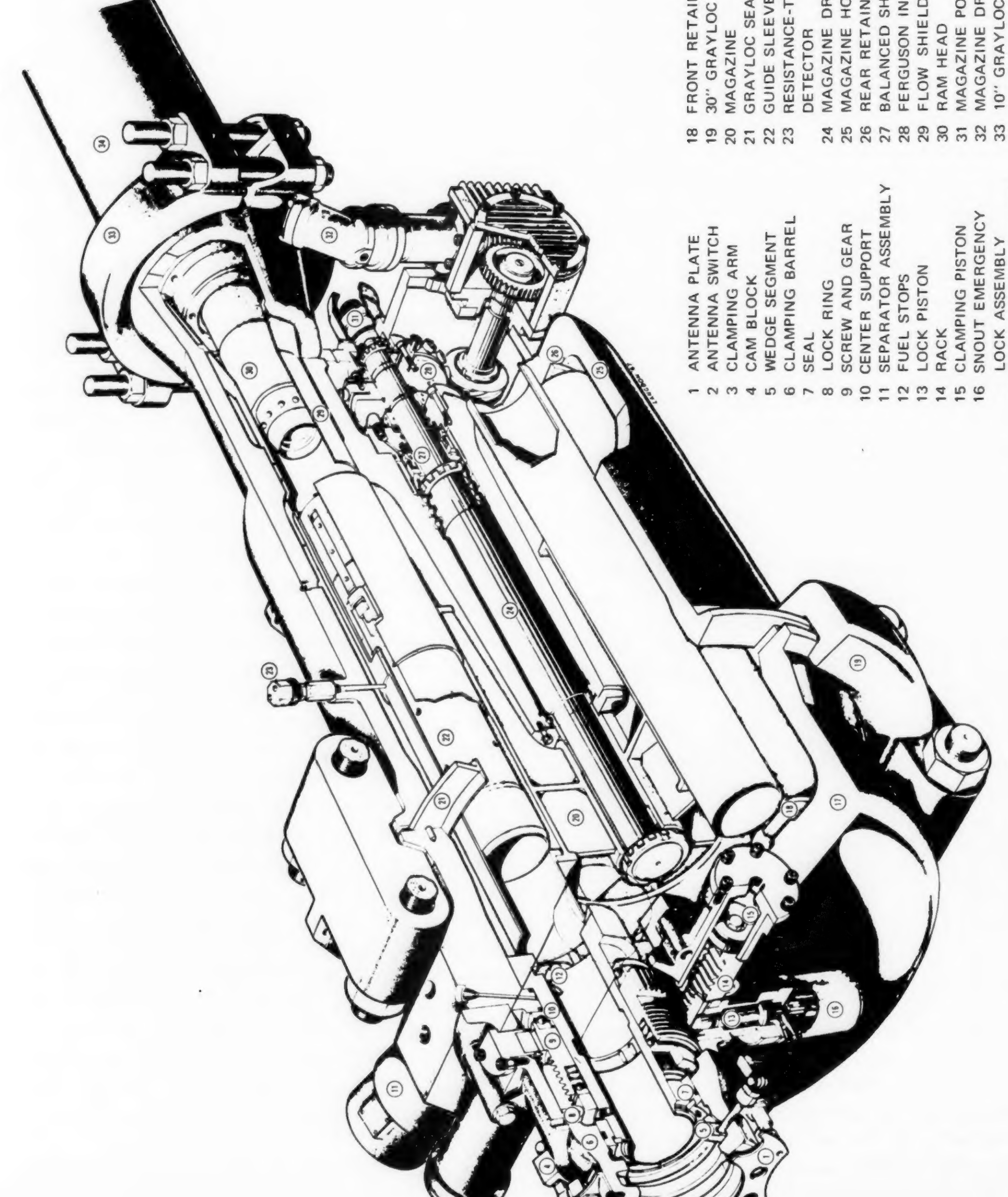
The transfer mechanism remains connected to the transfer tube (after the fueling-machine snout is sealed) until the internal temperature falls to 150°F. Its internal water level is then lowered slightly, and the transfer mechanism is aligned with the spent-fuel elevator. Spent-fuel bundles are positioned clear of the water inside the transfer mechanism and allowed to dry before being rammed onto the elevator. The elevator is lowered through a column of natural water to a horizontal duct that contains a wheeled conveyor for transporting the spent fuel to the storage bay.

MZFR Plant. The MZFR is a pressure-vessel-type reactor, using D_2O moderator and coolant and natural-uranium fuel. The thermal and electrical outputs are 200 Mw(t) and 50 Mw(e) (net), respectively. The reactor was designed and built by Siemens-Schuckertwerke A. G. and went critical in the fall of 1965.

The fuel-handling machine is mounted on top of the reactor in a vault inaccessible during reactor operation and is transported by a crane mounted above the reactor to connect it with the projecting fuel-channel end fitting. The complete fuel machine weighs about 30 tons and is heated through an external water circuit to 200°F to reduce the effects of thermal shock.

There are 121 fuel channels in the core, each containing two fuel bundles 188 cm long and 9.66 cm in diameter. (Seven other channels are also present and available for test purposes.) The fuel channels are closed off with closure plugs, each equipped with a bayonet latch that turns 120° to release the plug. The sealing member is a silver-plated hardened-steel ring, which is self-energizing from the internal pressure and seals against a hardened-steel wall of the stub end fitting. The two lengths of fuel are connected through a shield plug to the closure plug, and the entire assembly is removed as a single unit about 23 ft in overall length.

The fueling machine contains a five-station rotary magazine to carry spare strings of fuel and closure plugs. The magazine is rotated by a geneva motion drive running in water at the upper end of the machine. Each station in the magazine has a ram from which a



1 ANTENNA PLATE
2 ANTENNA SWITCH
3 CLAMPING ARM
4 CAM BLOCK
5 WEDGE SEGMENT
6 CLAMPING BARREL
7 SEAL
8 LOCK RING
9 SCREW AND GEAR
10 CENTER SUPPORT
11 SEPARATOR ASSEMBLY
12 FUEL STOPS
13 LOCK PISTON
14 RACK
15 CLAMPING PISTON
16 SNOUT EMERGENCY LOCK ASSEMBLY
17 MAGAZINE END COVER
18 FRONT RETAINING PLATE
19 30" GRAYLOC CLAMP
20 MAGAZINE
21 GRAYLOC SEAL RING
22 GUIDE SLEEVE
23 RESISTANCE-TEMPERATURE DETECTOR
24 MAGAZINE DRIVE SHAFT
25 MAGAZINE HOUSING
26 REAR RETAINING PLATE
27 BALANCED SHAFT SEAL
28 FERGUSON INDEXING DRIVE
29 FLOW SHIELD
30 RAM HEAD
31 MAGAZINE POSITION SWITCH
32 MAGAZINE DRIVE MOTOR
33 10" GRAYLOC CLAMP
34 RAM HOUSING

Fig. 3 Pickering fueling-machine head snout and magazine assembly.

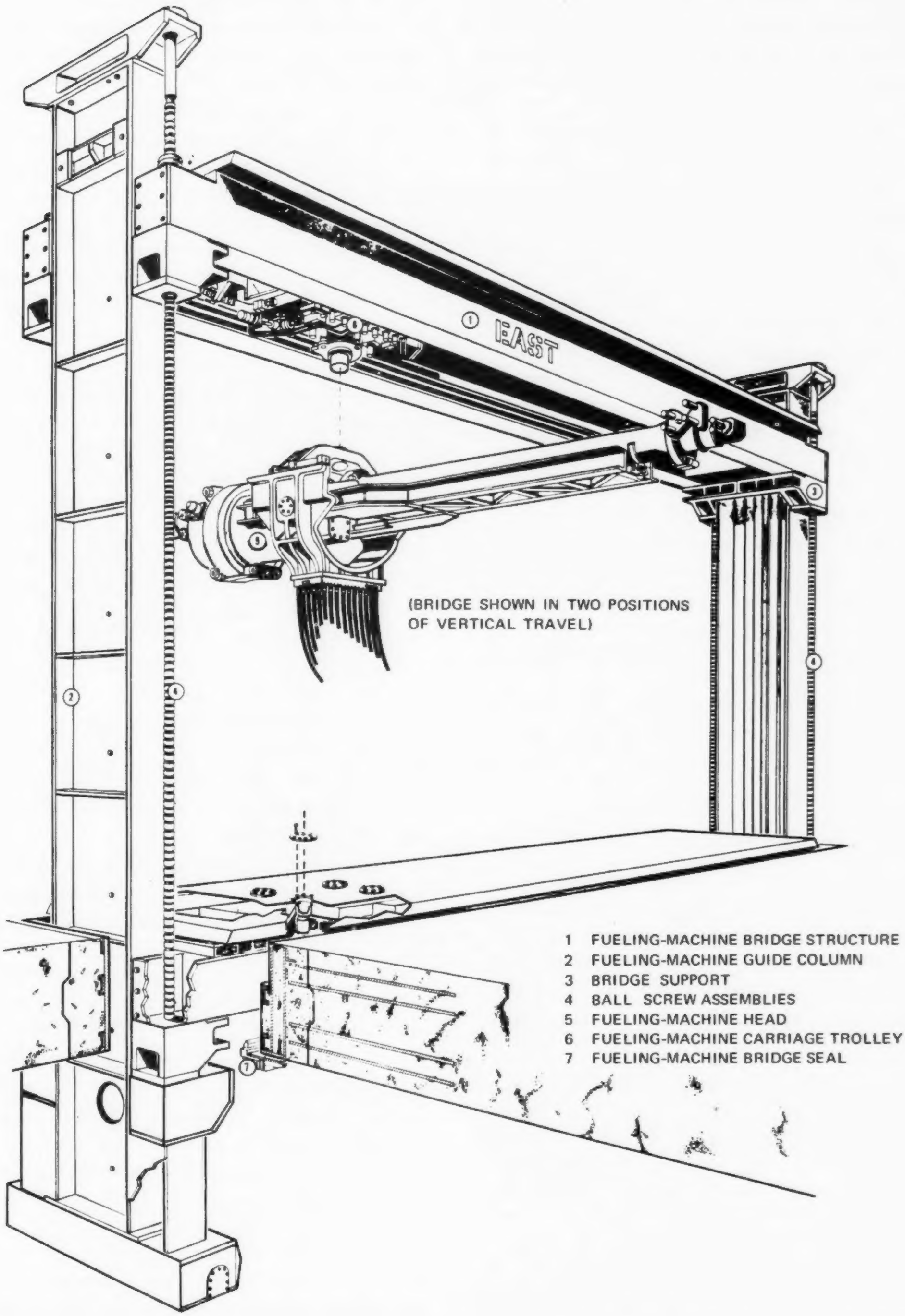


Fig. 4 Pickering fueling machine and support structure.

finger projects outward through a longitudinal slot. When a particular station in the magazine is rotated into position over a fuel channel, this finger engages a slot on the ball screw nut of the drive mechanism which provides the longitudinal ram motion.

The fueling sequence is as follows: two fuel bundles are coupled together and loaded by the fueling machine into a fuel channel. After partial exposure, the two bundles are withdrawn and inserted into the fuel shuffler that is located adjacent to the reactor. The relative positions of the fuel bundles are transposed, and they are rejoined and drawn back into the fueling machine. This fuel string is then inserted into a channel in a different section of the reactor, where it remains until it has achieved full burnup.

When the fuel is ready to be discharged from the reactor, the two-bundle fuel string is drawn up into the fueling machine and discharged vertically downward to a tubular tilting mechanism. This mechanism is then rotated to discharge the fuel through a downward-sloping transfer tube into the spent-fuel storage pool. The transfer tube contains isolation valves and also has a slot in the upper part of the tube through which a finger carried on a slowly moving chain projects to control the speed of descent of the fuel bundle to the storage pool.

The fuel is dried in the air lock by hot nitrogen gas so that the D_2O can be reclaimed. New fuel is similarly dried of H_2O when it is transported to the reactor from the storage bay.

Boiling Light-Water-Cooled Designs

Only two plants of this type have been designed: the Steam Generating Heavy Water Reactor (SGHWR) of the U. K. Atomic Energy Authority has been designed for on-load refueling, but an off-load system is being considered for some applications. The other plant, the Gentilly Nuclear Power Station, uses on-load refueling.

Winfrith SGHWR. In this prototype 100-Mw(e) unit, fission heat is transferred to the coolant in a series of parallel pressure tubes in a vertically oriented calandria. There are 104 Zircaloy-2 pressure tubes, each with a bore of 5.12 in. and about 13 ft overall length.

The fuel is in the form of bundles, 12 ft long, containing 36 core-length rods. Since light-water coolant is fed to the channels from the bottom, the fuel is loaded from above. The fuel-handling system is shown in Fig. 5. The fuel channels are extended beyond the fuel zone at both ends with stainless-steel pipes joined to the Zircaloy sections. The fuel channel is sealed at its upper end by a plug that carries a hollow support connected to a neutron shield plug. A detachable coupling at the lower end of the shield plug permits disconnection of the fuel string.

The fueling operations are performed over the top of a steel rotating shield structure, consisting of inner and outer rotating structures. These are eccentrically mounted so that any lattice position can be reached. A system of hydrostatic bearings was developed to overcome the problems of sealing and friction associated with the massive rotating structure, which weighs over 600 tons.

The fueling machine is enclosed within a stainless-steel vessel surrounded by a shield structure. The fueling-machine internal structure and nose unit are shown in Figs. 6 and 7. The mechanism is designed to permit positioning over, and sealing to, the standpipe extension of a fuel channel. A four-station magazine within the pressure vessel can accommodate one new boiling fuel stringer, one spent boiling fuel stringer, one spare seal plug for fuel-channel closure, and either a new or a spent superheat fuel stringer. A winch-operated hoist is used to withdraw fuel from the core.

Positioning of the fueling machine onto a standpipe is done under the visual control of an operator stationed on the top shielding; and final positioning is accomplished with a gimbal unit in the nosepiece. Whenever spent fuel is in the machine, however, the hoist and magazine motions are controlled from a central control room.

The spent fuel is disposed of by transferring it to a trolley-mounted tube, with which it is transferred to an underwater location. The mechanisms provide for sealing the tubes in which the fuel is carried, cooling the fuel, and lowering the tube pressure. The same devices are used for loading new fuel by reversing the operating procedures.

An off-load refueling system has been considered as an alternative to on-power refueling for a vertically oriented reactor. A pool of light water above the standpipes extending upward from the fuel channels would act as a shield. Connection with a standpipe could be made through an extension of it passing through a hole in the bottom of the pool. A simple hoist could then withdraw a fuel stringer into the pool. If the pool were built so that it extended laterally to the storage bay, then the hoist carriage could be used to transport the fuel stringer directly to the storage area. It is claimed that the overall economics of reactor operation using off-load refueling compares favorably

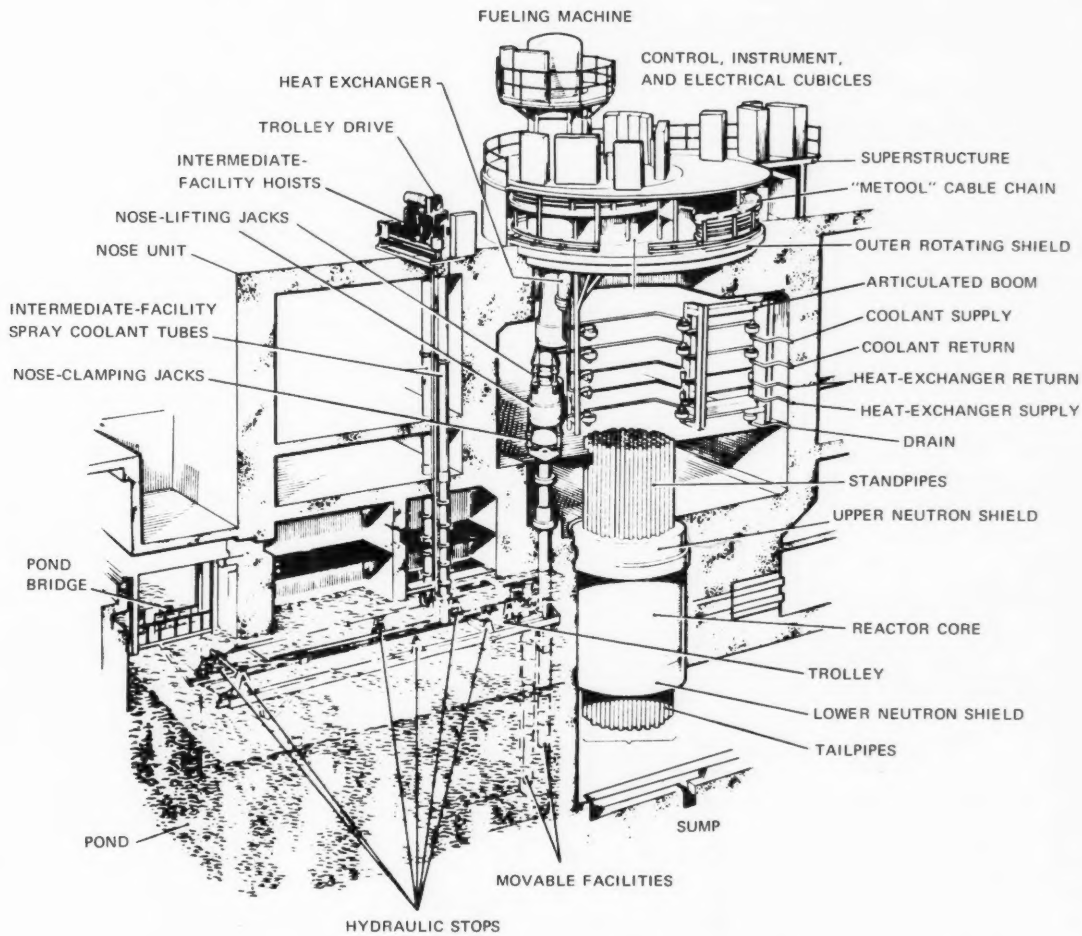


Fig. 5 Fuel-handling system for the Winfrith SGHWR.

with the more convenient, but more expensive, systems using on-power refueling.

Gentilly Nuclear Power Station. This 250-Mw(e) plant has been designed by AECL for Hydro-Quebec and features on-power refueling to allow for continuous power generation. The fuel-changing cycle is expected to take place over a 4-day period each week, leaving the remainder of each week for maintenance of the fueling machine should this be required.

Each channel contains 10 fuel bundles arranged in tandem. Normally only two of the bundles are replaced at any one time; thus a single pair of bundles can occupy a succession of up to five different axial positions in a channel prior to removal.

The reactor consists of a vertical cylinder containing 308 fuel channels. The fueling machine is located below the channels, and it is designed so that horizontal motions in two perpendicular directions can locate it directly beneath any one of the fuel channels. Vertical motion of the fueling-machine head allows it to be engaged to the fuel channel. After the fuel change is performed, the head is rotated until its axis is horizontal and moved to the horizontal transfer tubes. The fueling machine discharges the fuel strings from the reactor into one of a series of transfer tubes that run horizontally from the reactor vault to a water-filled trench. The spent-fuel string is cooled in the transfer tube to well below 200°F before being conveyed into the open water of the storage bay.

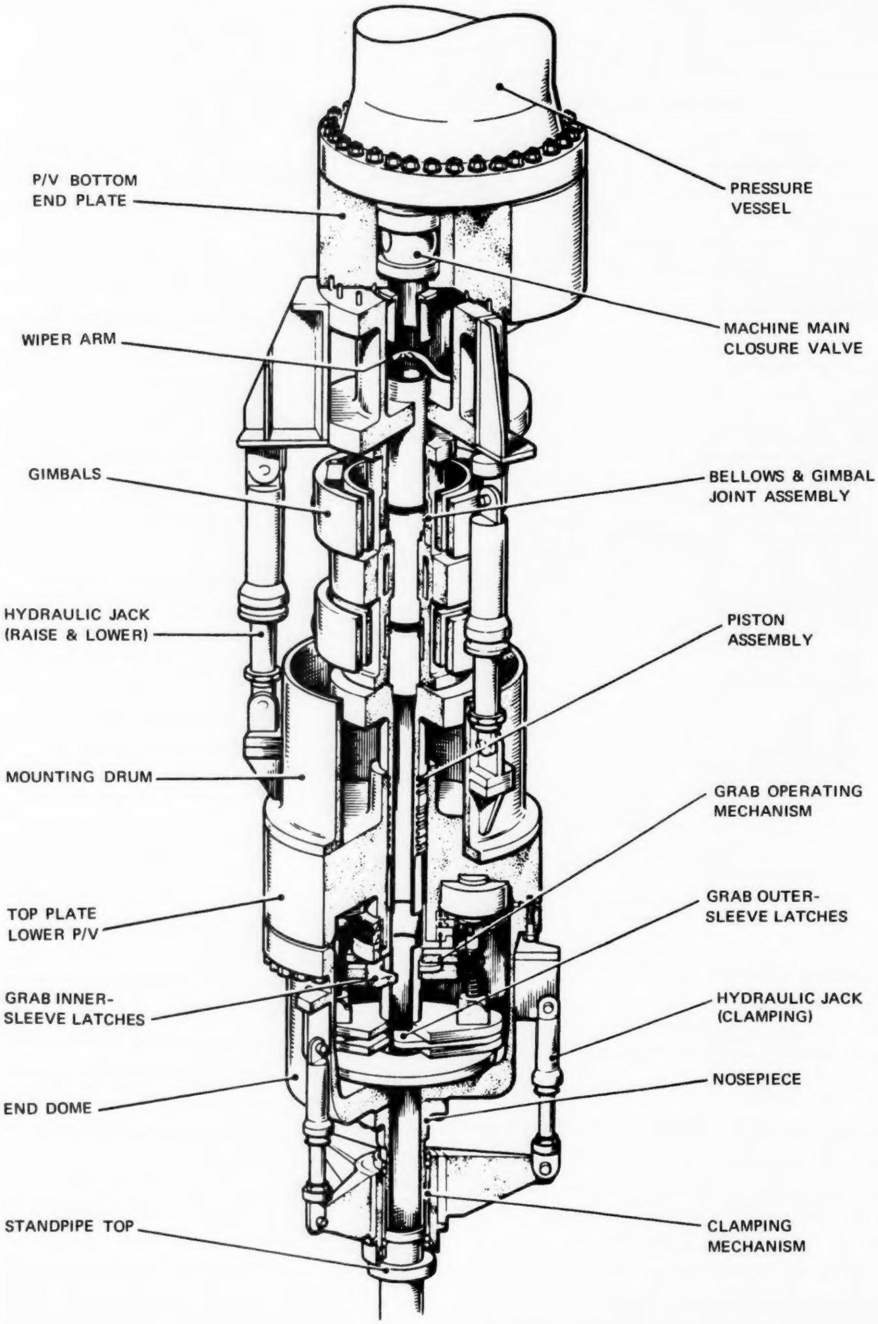


Fig. 6 Winfrith SGHWR refueling-machine internal structure.

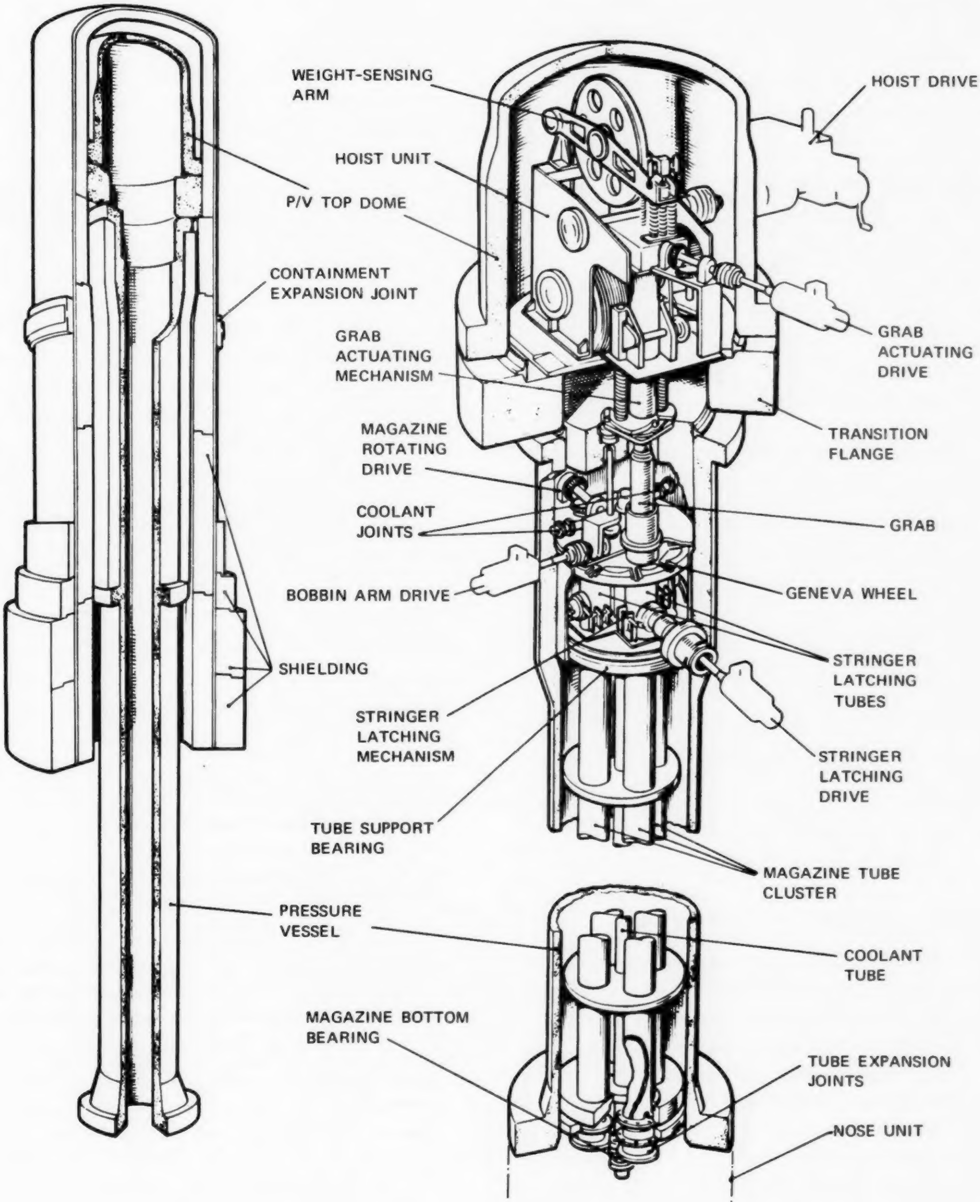


Fig. 7 Winfrith SGHWR nose unit.

When fuel first enters the fueling machine, it radiates heat at a rate of about 180 kw. The heat is removed by water circulated in an annulus between the lines in the fuel barrel and the casing. To safeguard the fuel cladding, workers always run the machine pressurized and hot (350°F), which ensures that the transition temperature of Zircaloy is avoided.

Fuel-Management Practices

In addition to the shuffling of fuel assemblies to different locations within the core to achieve larger fuel burnup, there are three other fuel-management practices which have an influence on the timing of the refueling (in plants that are refueled off-load), and, in the third case, also affect to a small extent the operations to be performed.

Premature Refueling or Reduced-Power Operation

Since electric-utility systems generally have winter- and/or summer-load peaks, the organization that operates the plant therefore usually schedules the plant shutdown during the spring or fall months. For achievement of this goal, particularly in the early years of plant operation when the performance and availability of the plant are likely to be somewhat erratic, it is possible that the utility will be required either to deliberately operate at reduced output for some period of time to conserve core life for use during the time of peak-load demand or to shut the plant down for refueling prior to the time corresponding to full core life so that the plant can be back on the line at full load during the peak-demand period.

End-of-Life "Coastdown" or "Stretch-Out"

When all control rods are fully withdrawn and all soluble poisons removed from the coolant in a pressurized-water reactor of either the light-water or heavy-water type, the reactor is capable of producing power at a decreasing rate over an extended period. The average coolant temperature decreases slowly with time, adding positive reactivity to offset the negative reactivity effects of fuel depletion. It is therefore practical, within limits, to remain on-line even at gradually declining capability and thereby defer the refueling shutdown to a more convenient period. This practice has been followed at the Yankee Nuclear Power Station.[7]

Cross-Shuffling

The transfer of partially used fuel from one reactor to a second that is just being started up offers the possibility of achieving a significant economic savings. This practice is a practical and economic consideration when starting up two reactors approximately 2 years apart. Although this requires careful scheduling, the potential savings are of the order of $4 million in the fuel cost of the second of two 1000-Mw(e) reactors. The basic approach is to run the first reactor on its normal first cycle and then to discharge half of the core and load it into the second reactor. At the next refueling, fresh fuel is loaded into each reactor. In effect, the second reactor is able to start operation at core isotopic-composition conditions that are closer to those which occur in its equilibrium cycle. This practice is presently contemplated in the case of at least three nuclear power stations: Peach Bottom Units 2 and 3 (BWR type); Browns Ferry Units 1, 2, and 3 (BWR type); and Zion Units 1 and 2 (PWR type).

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Reprocessing of the Elk River Reactor Fuel in the ITREC* Plant

By A. C. Schafer†

The initial core loading of the Elk River Reactor (ERR) will be reprocessed in the ITREC* plant of the Comitato Nazionale per l'Energia Nucleare (CNEN) located in southern Italy. Cold startup operations are presently under way in the plant, which has been designed to reprocess spent fuel elements and to remotely fabricate new fuel elements for a Th–U reactor under CNEN's Programma Ciclo Uranio-Torio (PCUT) program. It is the purpose of this report to give a brief description of the plant, the experience of the receipt of ERR fuel at the plant, and the plans for hot startup operations.

Part of the recovered ERR fuel will be reconstituted into a fuel loading for the heavy-water-moderated Halden reactor in Norway;[1] the remainder will be partitioned into uranium and thorium fractions. The uranium fraction will be converted to the oxide and returned to the AEC; the thorium fraction will be stored. Future plans for the ITREC plant include research and development for reprocessing and refabrication of HTGR fuels, and later it is expected to be used for developments related to reprocessing and refabrication of fast reactor fuels.

Historical

The preliminary design for the ITREC plant was initiated in 1960 by a group composed of Italian engineers from the CNEN organization and American engineers of the Allis-Chalmers Manufacturing Company. After completion of the preliminary design in April 1961, the detailed design was undertaken by Allis-Chalmers, with the Blaw-Knox Company as a subcontractor. This second phase was completed in July 1962 and was followed by a period of work in Italy by the CNEN organization for the design of general services for the research center located at Trisaia, near the town of Policoro, province of Matera, in southern Italy. The construction phase for the site and the ITREC plant lasted for 5 years, from 1963 to 1968. The plant was commissioned in the spring of 1968, and since that time efforts have been directed toward placing the plant in operation.

Plant-Design Basis[2-4]

The principal undertaking in the PCUT program is the extraction of fuel material from spent fuel elements of a Th–U reactor, the reconstitution of the extracted material in a usable form, and the assembly of new fuel elements for use in the same reactor or another reactor. Remote handling is provided for all of the processing steps, not only because of the fission-product activity but also because of the radioactivity associated with ^{232}U and ^{228}Th. Criticality control is by safe geometry in the chemical separations part of the plant and by batch limitations in the fuel-fabrication cell. Both the chemical and mechanical processes are carried out remotely from behind shielding walls.

The plant has been designed for an instantaneous capacity of 15 kg of Th–U oxides per day with enrichments up to 10% in ^{233}U. Minor additions to

*Impianto di Trattamento e Rifabbricazione di Elementi di Combustibile.

†Nuclear Associates International Corporation, Rockville, Md.

the plant, principally in the dissolving and recycle areas, should permit reprocessing at an instantaneous capacity of 30 kg of thorium–uranium per day.

The plant design departs from that of existing radiochemical plants in three ways:

1. The maintenance and service technique is based on rack-mounted units with quick-disconnect couplings.
2. Fuel-recovery and -refabrication operations are integrated in one plant.*
3. There is less fission-product decontamination of the recovered uranium and thorium. Only one cycle of solvent extraction is provided because remote fabrication is required in any case to protect against the gammas from ^{232}U. Fresh fuel is fabricated in a shielded cell adjacent to the reprocessing section of the plant.

Reprocessing and Refabrication Processes

The reprocessing sequence consists of four major phases: disassembly, recovery of the fuel material, reconstitution of the recovered fuel material, and assembly of a new element. The reprocessing sequence is represented schematically in Fig. 1.

Disassembly

The spent fuel elements are received in shielded shipping casks. Following surveys of the outside and inside of the cask and cooling of the contents, if necessary, the fuel elements are transferred underwater to fuel-storage racks in the pool. A remotely operated disassembly machine in the pool is used to remove the structural fittings but leaves all fuel-bearing rods intact so as not to expose the fuel material. The rods are loaded into a shielded magazine and carried from the canal to a chopping mechanism above the chemical-processing cell.

Functional tests of the disassembly machine have been completed, and a number of dummy ERR fuel elements have been dismantled remotely.

Recovery of Fuel Material

The chopping mechanism cuts the rods into short segments that fall into a dissolver, where the fuel material is dissolved using a concentrated nitric acid solution. The empty segments of stainless-steel tubing are left behind in a basket, which is subsequently removed and disposed of by burial.

The chopping mechanism has been subjected to functional tests, and it performs well so long as the fuel tubes are reasonably straight. Segments as short as 0.5 in. have been produced successfully with this mechanism. Further testing of the apparatus will be performed during the forthcoming cold runs.

The resulting nitric acid solution, containing fission products and the nitrates of thorium and uranium, is adjusted so that its concentration is suitable for the next step. The properly adjusted solution is then pumped into a mixer–settler device, where the thorium and uranium are separated from the fission products by solvent extraction using tributyl phosphate (TBP) and kerosene.

All equipment for feed adjustment, solvent extraction, and product concentration has been subjected to functional tests using water and nitric acid. Within the limitations of the tests, the equipment performed its intended functions as expected.

The major difficulties encountered have been associated with the transfer of fluids between vessels. The most serious of these has been a frequent plugging of the steam venturi of the jets. The majority of jets in the plant have a low capacity because of the small volumes of the vessels. The steam venturis in these jets have a minor diameter on the order of 1.2 mm, and it appears that foreign matter in the steam supply frequently plugs the venturi. Corrective action has been concentrated at providing a clean supply of steam to the jets. The steam is filtered in the main header, and the branches up to the control valves will be trapped to remove condensate and reduce corrosion.

The aqueous-product solution from the mixer–settlers is concentrated and then adjusted to ensure the proper Th–U ratio. Formic acid and urea are added to the adjusted solution, and then the mixture is heated to 60°C and passed through a bed of alumina pellets containing 5 wt.% platinum catalyst. This reduces the uranium valence state from +6 to +4 and permits precipitation of both uranium and thorium with oxalic acid. The resulting slurry of Th–U oxalates is centrifuged to remove the supernatant liquid.

The precipitation equipment was tested prior to its installation in the plant. It has operated well during these tests, but there is some concern that remote operation may present additional problems. Tests are scheduled for this summer and fall to establish the efficacy of the remote arrangement.

The Th–U oxalate solids are then passed through a calcining furnace, where the volatile materials are

*Experimental Breeder Reactor II, at the National Reactor Testing Station, integrates these two operations in a pyrochemical process.

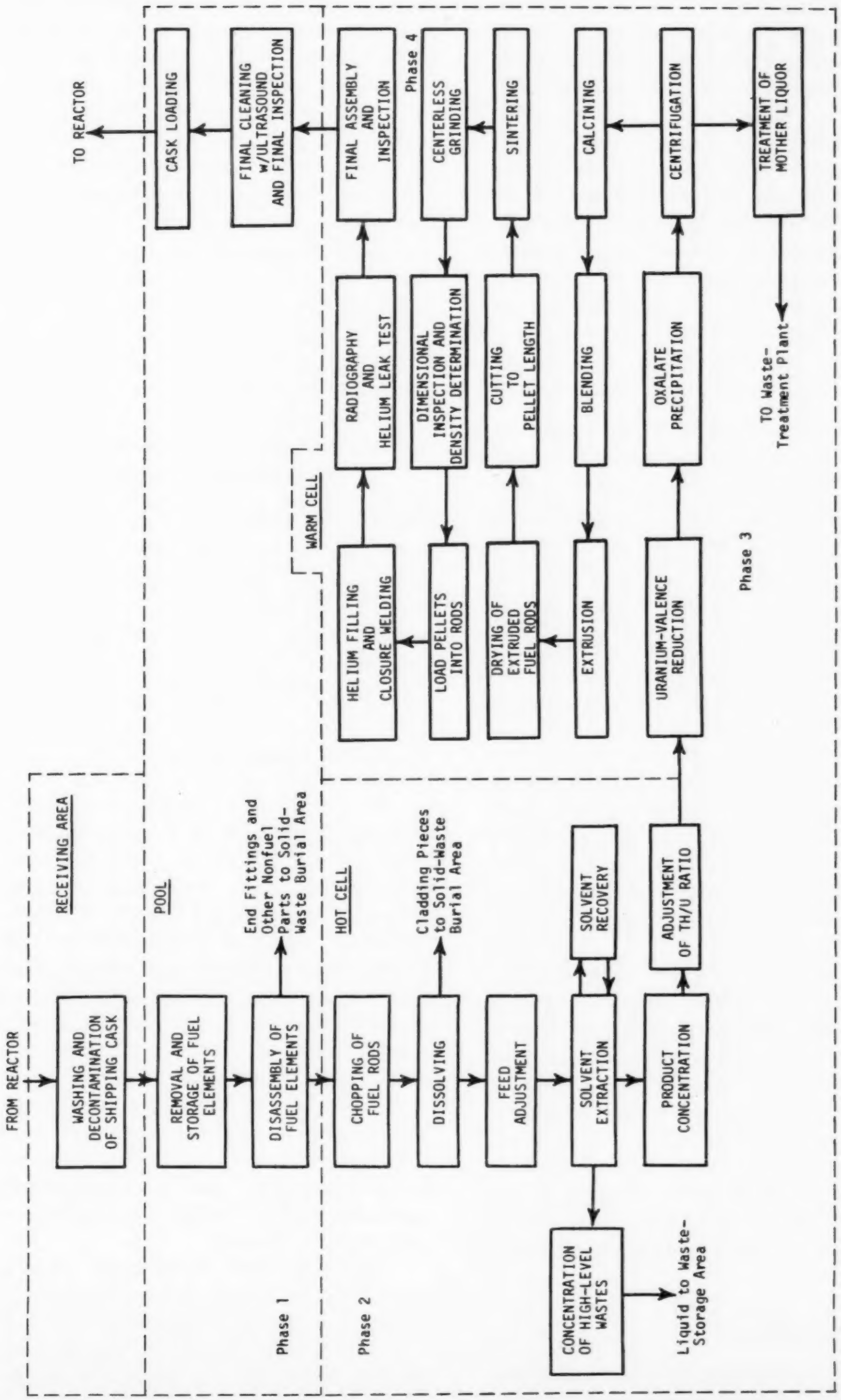

Fig. 1 Block diagram of PCUT process for RERR fuel.[1]

driven off and uranium and thorium oxides are formed. These oxides of uranium and thorium are the end product of the fuel-recovery phase. In the next phase, this material is formed into pellets for use in the fuel elements.

Functional tests of the continuous calcining furnace have revealed some difficulty in pushing the trays when the furnace temperature is at its operating value of 900°C. At this temperature the trays and the muffle are red hot, and the friction force, when the furnace is fully loaded, is sufficient to cause buckling between trays. Efforts are being directed toward reducing the friction between trays and muffle.

Reconstitution

The dry mixed oxides of uranium and thorium are passed through a "micropulverizer" to break up any aggregate bodies formed during the calcining step. The resulting oxide powders are blended with water and an organic binder in a Mixer–Müller prior to loading the resulting "clay" into an extrusion press. The moisture content of the clay is very important to the process, as is the size of the oxide particles. Both of these parameters are closely controlled by in-cell analyses.

The clay is extruded in 91-cm lengths, which are air-dried and cut to pellet length. The "green" pellets are then sintered at temperatures up to 1700°C to produce pellets of the required high density. The sintering furnace had not been tested at the time of this writing, but all other equipment had been tested both in-place and prior to installation. Pushing the trays through the sintering furnace when it is at operating temperature may prove to be a problem if the experience with the calcining furnace is any indication.

The sintered pellets will be ground to the specified diameter by use of a centerless grinder. The finished pellets pass through a gauging device that provides a continuous measurement of dimensions and continuous weighing of the finished pellets. Density calculations are obtained from a computer using these measurements as a continuous input.

This device has proved to be quite delicate in operation, and frequent repairs are required to the measuring deck. Efforts are being made to provide for remote replacement of some critical parts and strengthening of others.

Assembly of a New Element

After inspection, acceptable pellets are loaded into new fuel tubes. The length of the fuel stack is gauged, and then an insulating disk and spring are inserted. The end cap is partially inserted so that a helium filling hole in the end cap remains exposed. Prior to welding, the fuel tube is evacuated and filled with helium and the end cap is seated, covering the fill hole. The end cap is welded in a helium atmosphere. The closure welds are subjected to visual and radiographic inspections. Sample welds are subjected to a metallographic inspection for process control.

The welding machine has been tested both outside of the cell and in remote operation using Zircaloy tubes and end caps of the Recycle Elk River Reactor (RERR) design. Visual and metallographic inspections of the welds have shown good results, although process capability runs remain to be performed. The tests have indicated the advisability of a viewing device (e.g., a periscope) for use in checking the electrode position.

Completed fuel rods are subjected to a helium leak check as a final test of integrity. In the case of the RERR fuel, bundles of 25 rods are assembled and fitted with spacer clamps, after which the end pieces are pressed into position. Specially designed snap rings lock the rod bundles to the end pieces and obviate further remote welding, which is difficult. These machines have been tested and seem to present no particular operating problem.

Receipt of ERR Fuel

The first shipment of ERR fuel arrived early in December 1968 at the freight yards in Taranto, about 60 miles from the plant site. Upon notification of the arrival of the cask, health-physics personnel were dispatched to the freight yards to survey the exterior of the cask and the railroad car used for the shipment. Activity levels were found to be very low, indicating no damage or leakage from the cask. On the following day the cask was unloaded from the railway car, loaded on a lowboy, and transported to the plant site by truck. Traffic on the road was restricted, and a police escort was provided to the receiving area of the plant. The cask was then removed from the lowboy and placed in a dry pit within the containment building and adjacent to the pool area. The foregoing operations were performed without incident. Two minor complaints were voiced at this time:

1. No paper work accompanied the cask. Apparently customs officials had removed the papers attached to the cask for their records.

2. The shipper complained that the holes on the shipping frame did not correspond exactly to the drawings and templates previously provided.

The cask remained in the dry pit for a couple of weeks awaiting a license from the CNEN safety division to unload the cask. Measuring instruments were attached to the thermocouple connections on the exterior of the cask, and these showed no temperature rise above ambient. On December 23, preparations for cask unloading operations were started by taking an air sample from the interior of the cask. The pressure gauge attached to the cask vent read zero. It was noted that the gauge cover glass was cracked. The cask vent was connected to the chemical plant vessel off-gas system, and the drain connection of the cask was hooked to a low-pressure air-supply line. As air passed through the cask, it was sampled. The activity level of the initial sample was about 10^{-2} $\mu c/cm^3$, which is above the acceptable level of 10^{-5} $\mu c/cm^3$ required for opening the cask. When operations were resumed after the Christmas holidays, it was discovered that:

1. The pressure gauge attached to the cask was inoperative.
2. The rupture disk on the cask cover was corroded through, leaving an opening in the top of the cask.

After these deficiencies were corrected, the cask was flushed with air until the activity level met the unloading requirements. In the course of activities associated with the air purge, some water was detected at the lower drain fitting. The activity level of this water was 3 $\mu c/cm^3$ with a spectrum that covered the range of fission products. Following the air purge the cask was flushed with water that was subsequently stored.

On December 29 and 30, elements were unloaded from the cask and placed in individual containers in the storage pool. No further difficulties were encountered. The operations in the pool caused an increase in the activity of the ion-exchange resins in the pool cleanup system.

These resins were changed frequently to preclude radiation exposure to operating personnel because the columns were not shielded and resins were changed by direct handling. Planned modifications include shielding of the resin columns and provisions for remote removal and replacement of the resins. Health-physics coverage was excellent, and all operations were carried out without exposing personnel to appreciable radiation.

After a few months of storage, the fuel elements were removed from their individual containers. Samples of water taken from the containers showed a wide variation in activity between individual containers. According to AEC records, none of the ERR fuel elements shipped were classified as leakers.

Plans for Startup

The present plans for startup of the ITREC plant include the completion of functional tests and cold startup of the chemical portion of the plant. Hot startup is presently scheduled for the fall of 1969, dependent upon successful completion of the cold runs. The fuel-fabrication part of the plant will be put into operation in sequence with the chemical separations portion. Functional testing is presently under way.

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Pumps for Sodium-Cooled Fast Breeder Reactors

By Robert C. Allen, Sr.*

Although steam-cooled and helium-cooled systems are under survey for fast breeder reactors, sodium-cooled reactors are being developed most actively. Therefore pumps, with their bearings and seals, for sodium systems will be covered in this article. Some elements of seal design for the sodium systems are applicable to steam and helium systems.

The technical development of pumps thus far placed in service in sodium-cooled reactors has been outstanding, and continuing research and development should result in components that will meet the exacting reliability standards of electric utilities when used with the large [1000-Mw(e)] and greater plants of the future. This development will require, however, operating data on components to meet these high standards. Many of the published papers that deal with sodium pumps and shaft seals describe theoretical developments and laboratory tests of only short duration.

Pumps

Many types of sodium pumps have been built for fast breeder reactors, but only the following types are considered in this article:

• *Centrifugal pumps (including sealed-motor pumps).* These pumps are characterized by vertical shafts, a free sodium surface in the pump tank, a neutral gas blanket, and a shaft seal for cover-gas containment.

• *Electromagnetic pumps.* The principal types are d-c conduction (Faraday), a-c conduction, a-c flat linear induction, and a-c cylindrical linear induction (Einstein–Szilard).

Large sodium pumps are powered most conveniently by electric motors. Each plant requires comprehensive economic studies before the final drive can be selected. Some drives that have been employed are:

1. Direct-connected induction motors with variable-frequency source of electric power.

2. Wound-rotor motors with water rheostat control for variable speed.

3. Induction motors connected to pump shafts with electromagnetic couplings.

4. Wound-rotor motors with two sets of windings and complex switching systems to provide low-slip speed control corresponding to several synchronous speeds.

5. Geared vertical gas or steam turbines. The overall economics of such prime movers must be investigated before adoption. The effects on penetrations of reactor-containment buildings require careful consideration. Variable speed control is easily accomplished.

Centrifugal Pumps

The movement of sodium or its eutectic alloy with potassium, NaK, by a centrifugal pump does not present problems in fluid flow. Liquid sodium is about as fluid as water. Sodium, from its melting point to 752°F, covers the same range of viscosity values as

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water between 104°F and its boiling point.[1] The specific weights for sodium at various temperatures are:

Temp., °F	Specific weight, lb/cu ft
300	57.2
500	55.5
700	53.8
900	52.0

Designers of centrifugal pumps are confident that the present mass of fluid-flow data acquired over many years of experience with centrifugal water pumps can be applied directly to design of sodium pumps without pilot model tests. The fluid-flow design of the passages in the centrifugal-pump impeller, the entry, the discharge diffuser, and the outlet casing is beyond the scope of this article. These design aspects are covered in Refs. 2 and 3.

Centrifugal pumps have been in operation at the EBR-II reactor at the National Reactor Testing Station, Idaho Falls, Idaho; the Enrico Fermi reactor, Monroe, Mich.; and the Rapsodie reactor, Cadarache, France.

The motor-driven centrifugal sodium-coolant pump used with the EBR-II reactor (Fig. 1) is vertically mounted in the main sodium tank to provide a free sodium surface in the gastight pump enclosure. The upper seal confines the cover gas and the small amount of sodium vapor it contains. A motor-generator set with an eddy-current coupling provides the variable frequency and voltage for the induction motor driving the pump. The gastight motor enclosure contains the argon cover gas. This simplifies the shaft-seal problem. A lower labyrinth seal retards diffusion of sodium vapor into the motor enclosure. The output of each pump is approximately 5000 gal/min at a sodium temperature of 700°F. The pump efficiency at this output is 0.80. The operating speed range is 10 to 100% of normal.

The primary sodium pump for the Fermi reactor (Fig. 2) is so designed that the pump can be lifted vertically into an argon-filled plastic bag for inspection or repair while the argon cover-gas blanket is maintained above the sodium level. The check valve that prevents backflow of sodium when a pump is shut down is part of the removable assembly. A sodium-filled dashpot in the check valve prevents violent shock when the valve is closed. The driving motors are of the three-phase wound-rotor type. Speed is changed with

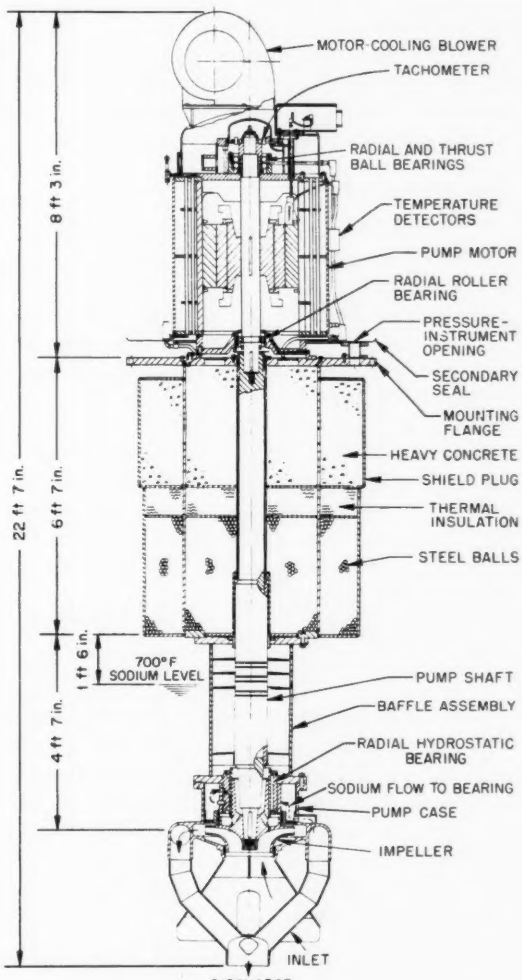

Fig. 1 EBR-II primary sodium pump. (From page 172 of Ref. 4; this figure appeared originally in USAEC Report ANL-5719, 1962.)

liquid rheostats. Each of the three primary sodium pumps is rated as follows:

Capacity, gal/min	10,000
Total head, ft	310
Sodium temp., °F	550
Pump efficiency	0.77

These figures apply to a reactor output of 300 Mw(t) with three pumps operating.

The sodium pump for the Rapsodie reactor (Fig. 3) follows the same design pattern as the preceding examples. Some important features are:

• Two-bearing hollow pump shaft.
• Upper bearing in accessible location.
• Coupling above upper pump-shaft bearing, which greatly facilitates determination of alignment.
• Means for vortex prevention.

Sealed-Motor Centrifugal Pumps

The induction driving motor of one sealed-motor pump (Fig. 4) is arranged with a stationary stainless-steel cylindrical "can" in the bore of the stator to prevent sodium reaching the windings. In one version of this design, hydrostatic bearings are supported by sodium from the pump discharge. The selection of bearing materials is important because the bearings must support the shaft without abrasion or objectionable wear when the unit is started and brought to operating speed. Temperatures in the sodium should not exceed about 575°F. Sealed-motor pumps have not been employed extensively for liquid metals; their chief applications in this service have been for small service pumps.

Status

Present design limit for the size of primary sodium pumps is 120,000 gal/min, with 450 ft total head. At 85% efficiency the driving motor must deliver 14,000 hp to the pump shaft. Pump designers would prefer a single-stage pump for these conditions, but this decision must be delayed until detailed design data are available.

Electromagnetic Pumps

The basic principle of the Faraday d-c conduction pump is shown in Fig. 5. When a magnetic flux passes through the sodium conduit and a large low-voltage electric current is applied through the flow at right angles, a propulsive thrust is imposed on the liquid metal. The compensating arrangement of electrical leads for the d-c Faraday pump (Fig. 6) minimizes the effect on the magnetic field of the flow of current across the pump duct. Tapering the pump duct in the direction of flow has the same compensating effects.[1]

Magnetic flowmeters are an important application of the d-c Faraday pump principle, in which the device becomes a d-c generator. A conduit of conducting fluid, moved mechanically and passed through a magnetic field, generates an electric current at right angles to the impressed magnetic flux.[1] Even for large pumps, overall efficiencies exceeding 40% are not predicted when current supply-source and bus losses are included.[5]

Alternating-current conduction pumps have been employed only in small sizes because of the low efficiencies and are similar to the d-c Faraday pumps. However, the alternating magnetic field passing

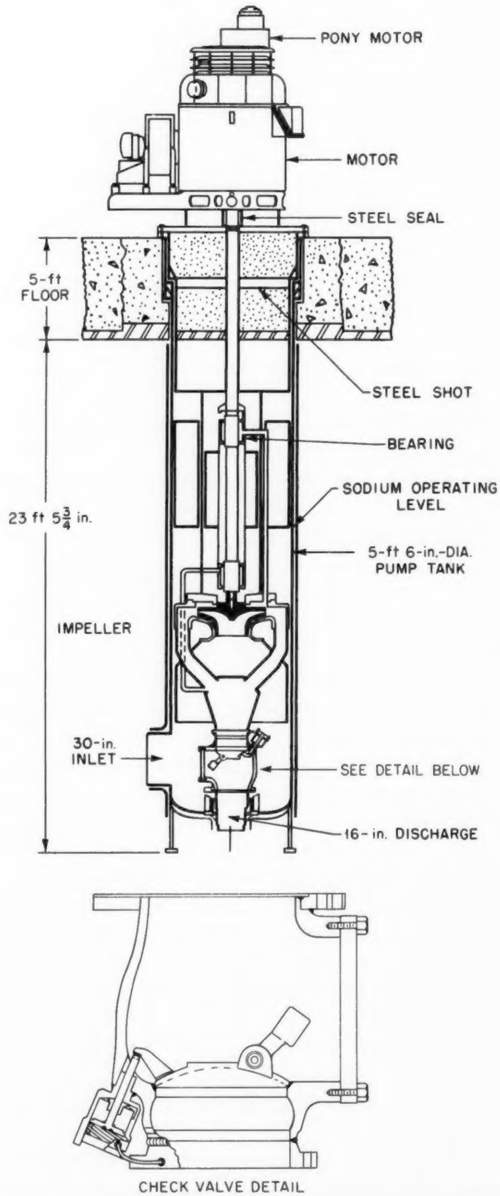

Fig. 2 Fermi primary sodium pump. (From page 176 of Ref. 4; this figure appeared originally in Enrico Fermi Atomic Power Plant Technical Information and Hazards Summary Report, 1962.)

through the pump duct must reverse synchronously with the currents induced in the fluid.[5] A large low-voltage current is supplied by a stepdown transformer, and the low-voltage secondary winding is connected to the narrow opposite sides of the flattened pump duct. The magnetic circuit of the transformer is arranged with poles at the opposite wide sides, with the flux passing through the duct at right angles to the current. Use of this pump for large liquid-metal flows is not anticipated.

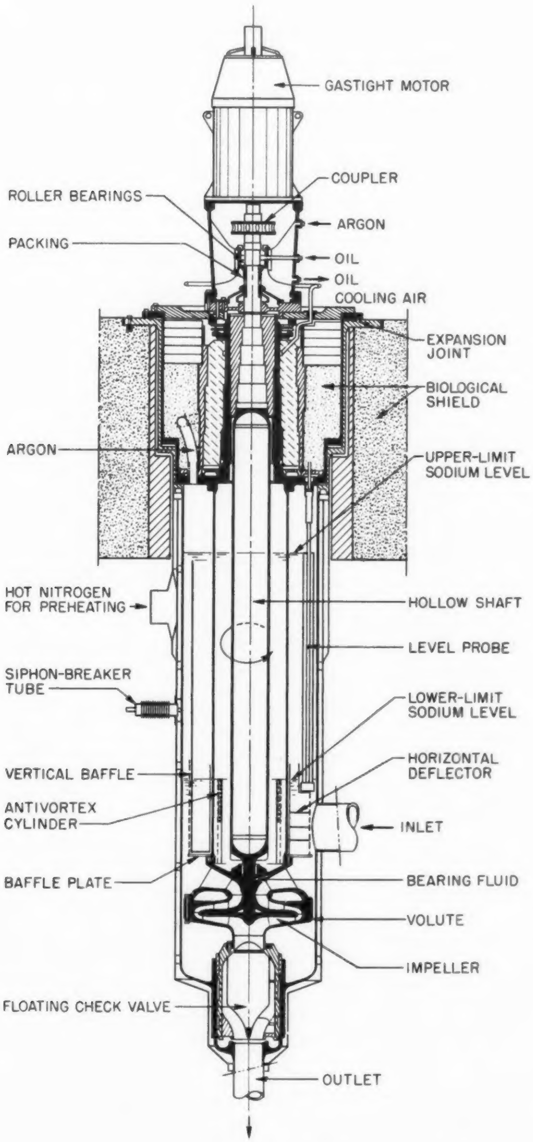

Fig. 3 Rapsodie primary sodium pump. (From page 744 of Ref. 4; this figure appeared originally in C. P. Zaleski and L. Vautrey, Le Reacteur Rapide Surregenerateur, Vols. 1 and 2, French Report CEA-2193, Commissariat à l'Energie Atomique, France, 1961.)

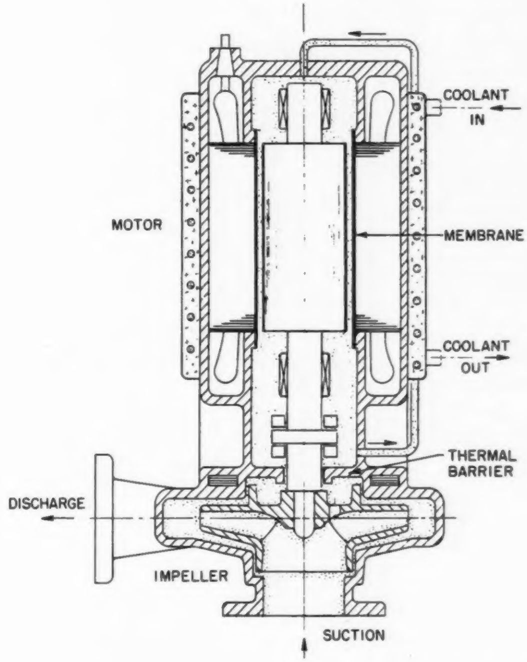

Fig. 4 Sealed-motor pump. (From page 168 of Ref. 4; this figure appeared originally in *Liquid-Metals Handbook*, 1955.)

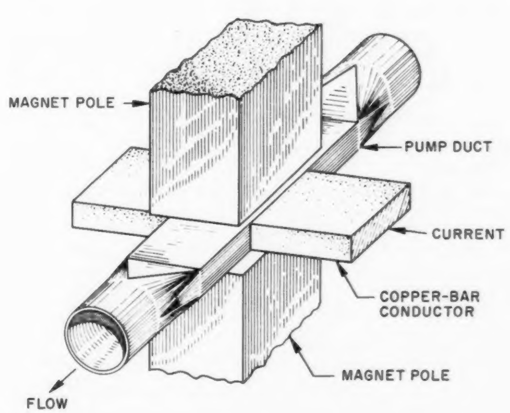

Fig. 5 Faraday d-c electromagnetic pump. (From page 179 of Ref. 4; this figure appeared originally in *Liquid-Metals Handbook*, 1955.)

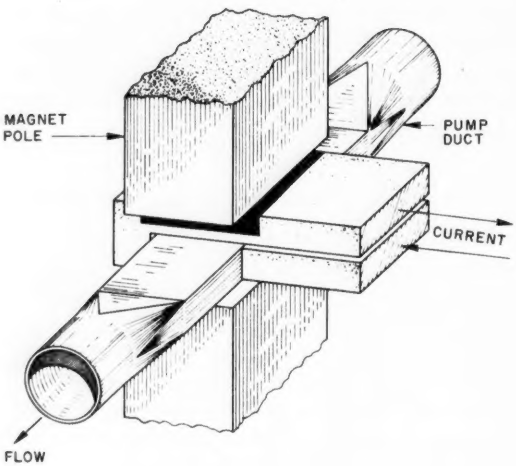

Fig. 6 Faraday pump. Leads arranged for field compensation. (From page 179 of Ref. 4; this figure appeared originally in *Liquid-Metals Handbook,* 1955.)

A further application of the a-c version is the linear induction pump. A three-phase alternating current supplied to suitable windings causes circulating currents in the sodium along a flattened conduit which generate a propulsive thrust on the liquid metal. This system is employed for the Dounreay reactor, in northern Scotland. Dounreay's coolant flow (Fig. 7) is divided into 24 primary circuits, each with an a-c linear induction pump. The efficiency of these pumps[4] is said to be about 30%.

The flattened tube for the fluid path is a design problem common to almost all types of electromagnetic pumps. The tube material must be as thin as possible and have low electrical conductivity. Magnetic

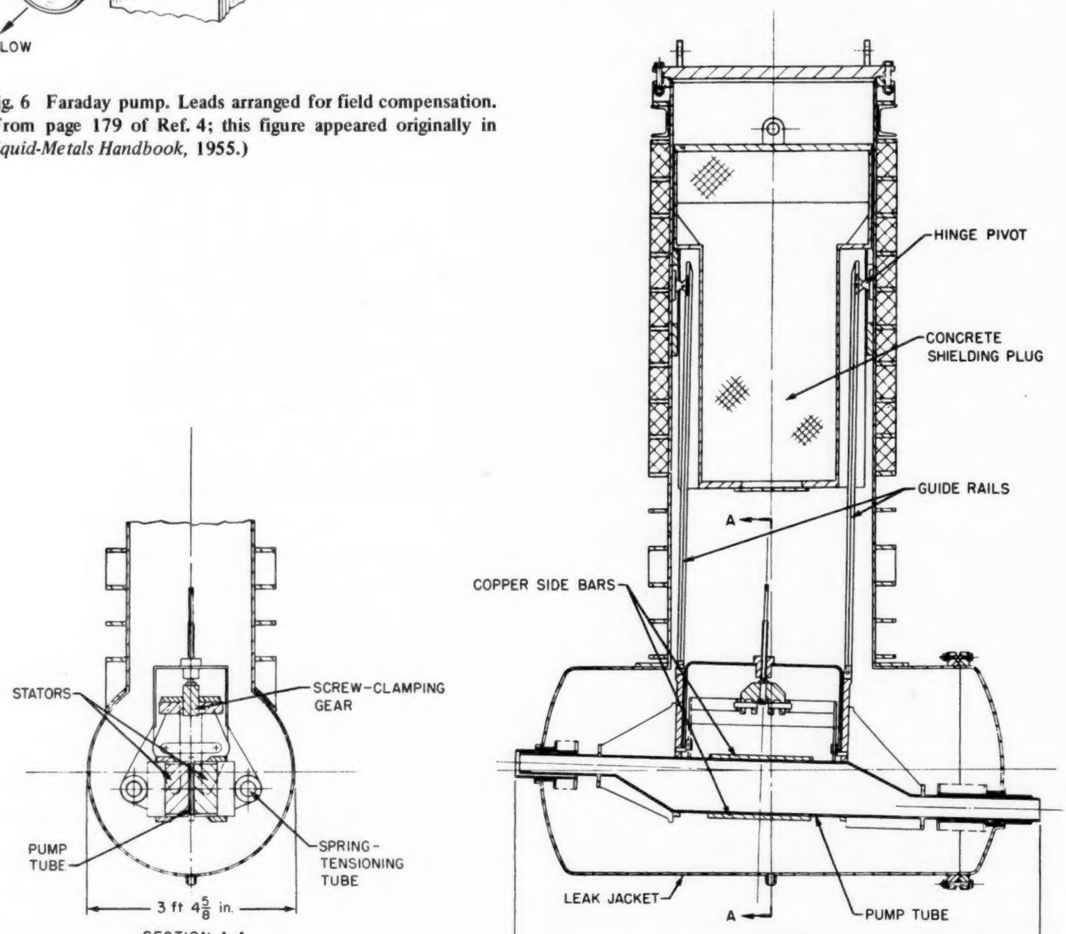

Fig. 7 Dounreay a-c linear induction pump. (From page 183 of Ref. 4; this figure appeared originally in R. R. Mathews et al., Proceedings of the Symposium on the Dounreay Fast Reactor, London, Dec. 7, 1960, Institution of Mechanical Engineers, London, 1961.)

pumps require high-ampere currents of low voltage, which involve either generating low-voltage direct current or transformer losses for the a-c systems. The annular flow section of the Einstein–Szilard pump avoids the design difficulties with the flattened tube of the conventional linear induction type. However, adoption for high outputs is unlikely because it has lower efficiency than centrifugal pumps.

Although the electromagnetic pump is less efficient than centifugal pumps, it has the great advantage of having no moving parts and no shaft seals.

Bearings

The bearings considered for sodium-cooled reactors are the fluid-piston (hydrostatic) and the pivoted-pad types.

Hydrostatic or Fluid-Piston Type

The fluid-piston bearing (Fig. 8) has been developed for equipment that must operate entirely with sodium. For hydraulic compensation to oppose shaft displacements, the sodium enters each of the four fluid-support chambers through a metering orifice. The pressure of the sodium is enough to form a hydrostatic supporting area separating the metal shaft from the containment rim around the bearing periphery. The radial clearance permits a nominal leakage flow over the containment rim. A displacement of the shaft increases the supporting pressure on the side at which the radial clearance is reduced, with a lowering of supporting pressure on the opposite side of the shaft. This pressure compensation keeps the metals apart. Equal transverse pressures keep the shaft centered in the bearing.

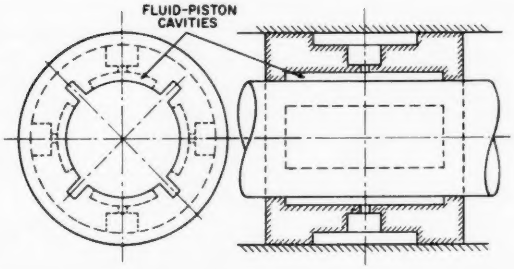

Fig. 8 Hydrostatic support bearing of the rigid-support, hydraulic piston type. (From Allis-Chalmers Mfg. Co., Final Report 43-112, Development of Hermetically Sealed Centrifugal Pump Units for Liquid Metals, 1953.)

Increased capacities and longer driving shafts between the upper operating level and the lower pump bearing create the possibility that the pump casing will be distorted by the expansion of the large sodium piping. This possibility makes a spherically mounted lower bearing desirable. This will permit angular movements and prevent the rims of the fluid containments from cutting through the supporting sodium, which could score rims and increase radial clearances.

One solution to this problem (Fig. 9) divides each of the four supporting areas shown in Fig. 8 in half and

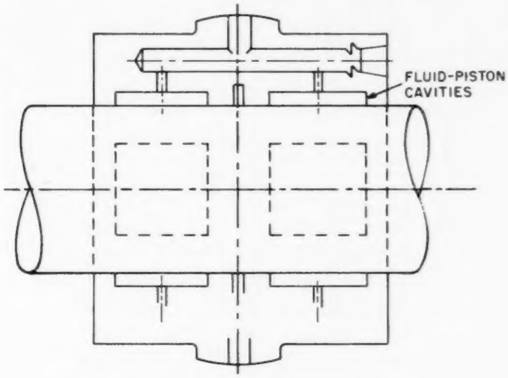

Fig. 9 Hydrostatic support bearing of the self-aligning type.

provides a metered quantity of sodium to each of the eight supporting cavities. The increase in leakage at diagonally opposite ends of the bearing and reduction in clearances at diagonally opposite zones react to center the bearing with the shaft. Thus the bearing accommodates changes in angular alignment without metallic contacts.

Pivoted-Pad Bearings

A pivoted-pad bearing (Fig. 10) provides complete hydrostatic support at low surface speeds, or hydrodynamic support with a suitable lubricant at an appropriate surface speed. The lubricant, either a gas or a liquid, is introduced to the bearing pad by a flexible tube.[6] A predetermined amount of supporting fluid enters each of the four support areas through a metering orifice at a pressure sufficient to prevent metallic contact between bearing and shaft, even at zero speed. Details of the supporting structure are omitted in the figure. An interesting application is the support of the north axis of the Mt. Palomar telescope,[7] where the load is 500,000 lb. The turning

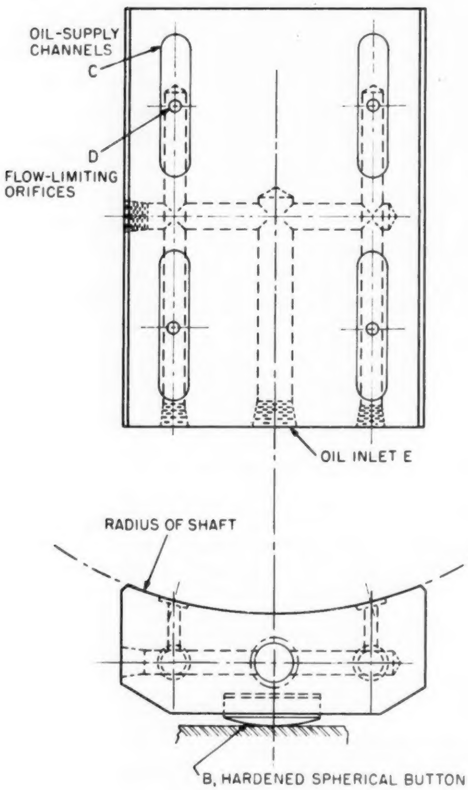

Fig. 10 **Tilting-pad support bearing.**

moment of the hour axis was reduced to about 1/600th of the value that would have existed if roller bearings had been used.

The lubricating properties of sodium in rotating-shaft bearings are not well established. Good hydrodynamic lubrication for rotating shafts requires maximum adhesion and minimum cohesion.[8] The variable wetting properties of sodium in hydrodynamic film bearings require special attention.[1,9]

Development of hydrodynamic bearings for sodium must be continued. Reducing bypassed sodium is a desirable objective. Tilting pads are considered necessary for hydrodynamic bearings with sodium lubricant.

Seals

The demands of electric utilities for extreme reliability of all components for base-load service require a careful review of design of sodium-pump shaft seals. Future pumps may have capacities requiring at least 14,000 hp and correspondingly larger diameter shafts. However, the speeds of rotation will be progressively lower for the larger outputs. The Fermi 10,000 gal/min primary sodium pumps have a shaft diameter of 8 in. at the upper gas seal, and the maximum speed of rotation is nearly 900 rpm.

Face-Type Seals

There has been more service experience with face-type shaft seals than with other types. In applications of these seals, the volume of the cover-gas space must be great enough so that the sodium level can vary, without sodium reaching the shaft seal. The seal also must prevent leakage of contaminated argon to the atmosphere. In the upper gas seal of the Fermi primary sodium pumps (Fig. 11), there are two face seals in each shaft assembly. Multiple springs hold the axially sliding faces on the shaft in contact with the stationary seal faces and hold matching seal faces together to contain the cover gas when the pump is shut down. Also, the lubricant-throwing slinger and the stationary dam (Fig. 11) are provided to prevent lubricant leakage from reaching the lower sodium space.

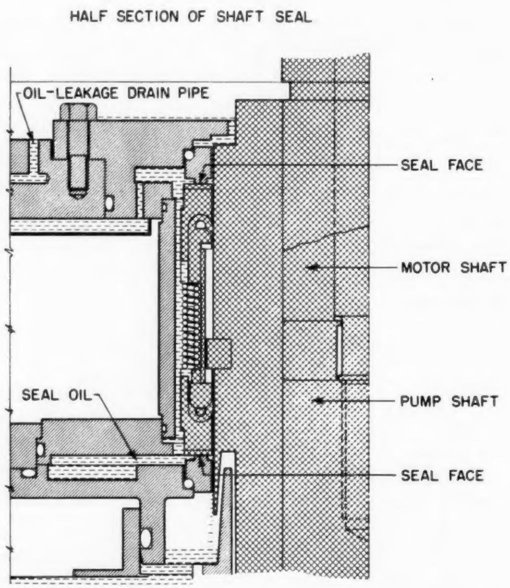

Fig. 11 **Face seal for primary sodium pumps of the Fermi plant. ▨, stationary; ▩, rotary; ▤, oil. (From page 178 of Ref. 4; this figure appeared originally in Personnel Training Program Plant Manual, Power Reactor Development Company, 1960.)**

Development of Face Seals

Figure 12 illustrates the design problems that must be solved to develop a face seal for long service life. Salient features include:

1. Rotating collars on the shaft with very light press fits to minimize distortion during assembly. Driving keys, which are essential, are not shown.
2. Rotating collars and stationary seal rings with plane faces, permitting surfaces to be lapped and tested with optical flats.
3. Closely spaced springs arranged to maintain contact between nonrotating members and rotating collars at their respective seal faces. The total spring load should be the minimum necessary to ensure adequate fluid film thickness at all times when the pump is operating and to provide a gastight seal when the shaft is not rotating. Spring pressure should be enough to prevent excessive lubricant flow at maximum speed.
4. A nonrotating seal-ring assembly arranged to slide easily in a vertical direction as required by shaft expansion.
5. Upper and lower stationary seal rings prevented from rotating by four equally spaced radial keys in each ring assembly. The short vertical dimensions of the integral key members permit minor angular errors between the axis of the seal housing and that of the shaft. Adequate radial and circumferential clearances must be provided for each integral key.

The figure does not show provision for draining lubricant that escapes from the lower seal ring or removing it from the atmospheric space above the upper seal ring.

Whatever materials are chosen, the matching surfaces of a face seal must be positively lubricated, either with a liquid or a gas, if the seal is to survive over long service periods. Therefore reliable means must be provided to ensure that the seal faces will be separated by lubricant films under all conditions of operation. The lubricant films must be thick enough to avoid objectionable heating. The seal also must effectively prevent the escape of cover gas to the atmosphere during shutdown periods.

The introduction of lubricant to the seal faces through axial holes in the nonrotating rings is shown in Fig. 12. Tests under simulated service conditions will be necessary to determine the specific design details for lubricating in this manner.

One method for accelerating the flow of lubricant between the seal faces (Fig. 12) uses spiral grooves in the stationary ring that can be produced by chemical etching or by machining.[10,11] Gas seals have also been tested with spiral grooves in the rotating faces that adjoin flat stationary faces. Before adoption, the best forms and locations of lubricant grooves must be determined by research and development. The radial widths of the seal-ring faces shown may be greater than necessary. The ultimate proportions also must be determined experimentally.

In tests of experimental face seals, temperature differences caused seal-ring faces to assume curved figures of revolution, which gave greater axial clearances at the outer peripheries of the seals. This difficulty might be partly offset by directing the lubricant flow so that nearly uniform temperatures will be maintained over front and back surfaces of the rotating seal rings. The use of a material of high thermal conductivity should reduce the distortion of the stationary rings. This thermal distortion must be carefully analyzed (Fig. 6 in Ref. 10).

The seal system for a liquid lubricant should include a drain tank in which the normal liquid level is at least 2 ft below the lower dam that prevents leakage of the lubricant into the cover-gas space above the sodium. The drain tank must have enough free surface to permit dispersion of any gas bubbles formed by the action of the seal.

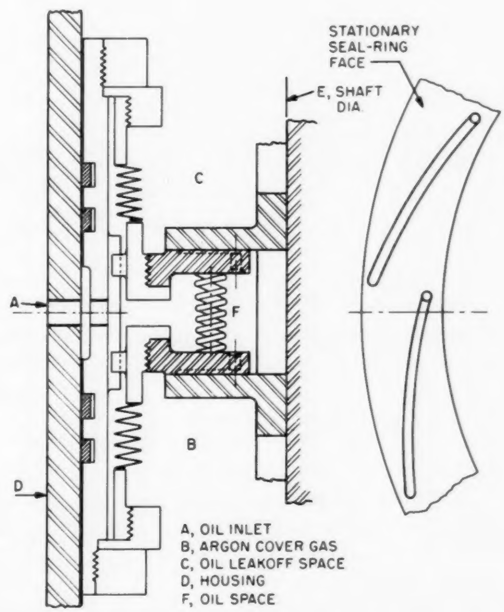

Fig. 12 Tentative design for vertical sodium-pump face seal of the liquid-lubricant type.

Supplemental Labyrinth Seals

Figure 13 shows an overall seal assembly in which the upper seal is of the face type shown in Fig. 12 and is supplied with liquid lubricant. Shown is a lower labyrinth seal intended to retard the flow of sodium vapor to the argon space below the upper liquid seal. Argon can be introduced between the two groups of labyrinth seals at a pressure slightly higher than the cover-gas pressure above the sodium. Selecting adequate length of labyrinth seal, radial clearances, and pressure differential can effectively reduce the upward diffusion of sodium vapor.

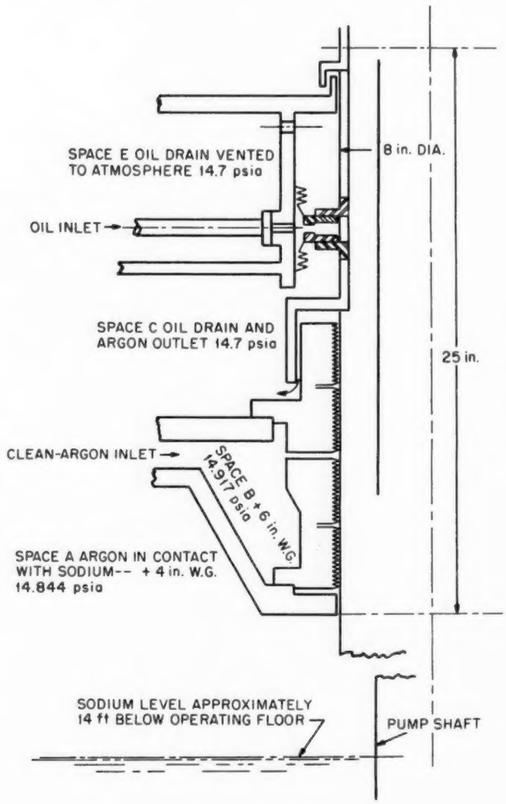

Fig. 13 Combined labyrinth and face seal.

If rigidly mounted, the lower labyrinth seal can cause objectionable local heating of the shaft if relative movement changes the radial clearance enough to cause rubbing contact.

Labyrinth seal rings (Fig. 14), made in segments with springs to hold them against shoulders in the containment housing, have been used in large steam turbines. This construction greatly reduces the intensity of an accidental rub.

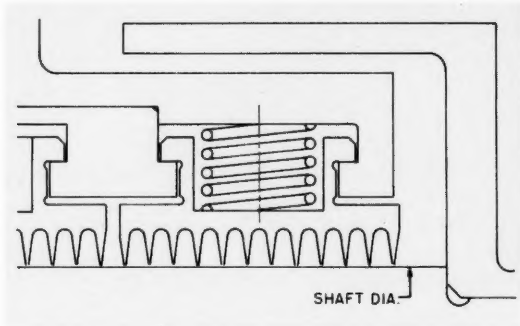

Fig. 14 Labyrinth seal of the steam-turbine type.

Gas-Supported Face Seals

Some of the design logic for liquid-lubricated seals applies to gas seals.

With the gas-supported seal, unit pressures must not exceed a few pounds per square inch if hydrodynamic lubrication is employed. The thin fluid films in gas seals (about 0.0001 in.) require filtering the gas supply to prevent the admission of solid impurities. Further experimental work must be carried out on face-type gas seals to establish a design with the required high order of reliability.

During the Pratt & Whitney research program on face seals,[12] one seal was found to have an unexpectedly long life, although previous test seals thought to be identical had limited life cycles. Working with optical flats, technicians found that the exceptional seal ring had warped to an undulating curvature with three valleys in the circumferential direction, about five helium light bands deep, corresponding to a depth of about 0.000055 in. A special lapping device was designed to lap a new seal ring to approximately duplicate the exceptional ring. Excellent performance was repeated. The effect described was found with gas-supported seals, but there should be a corresponding advantage with liquid lubricants.

Floating Bushing Seals with Liquid Lubricant

A good background of experience with liquid-lubricated floating bushing seals (Fig. 15) indicates that the floating bushings must be positioned so they are free to follow the shaft without requiring objectionable reaction forces to compensate for either radial or angular shaft misalignment; hence the spherical aligning

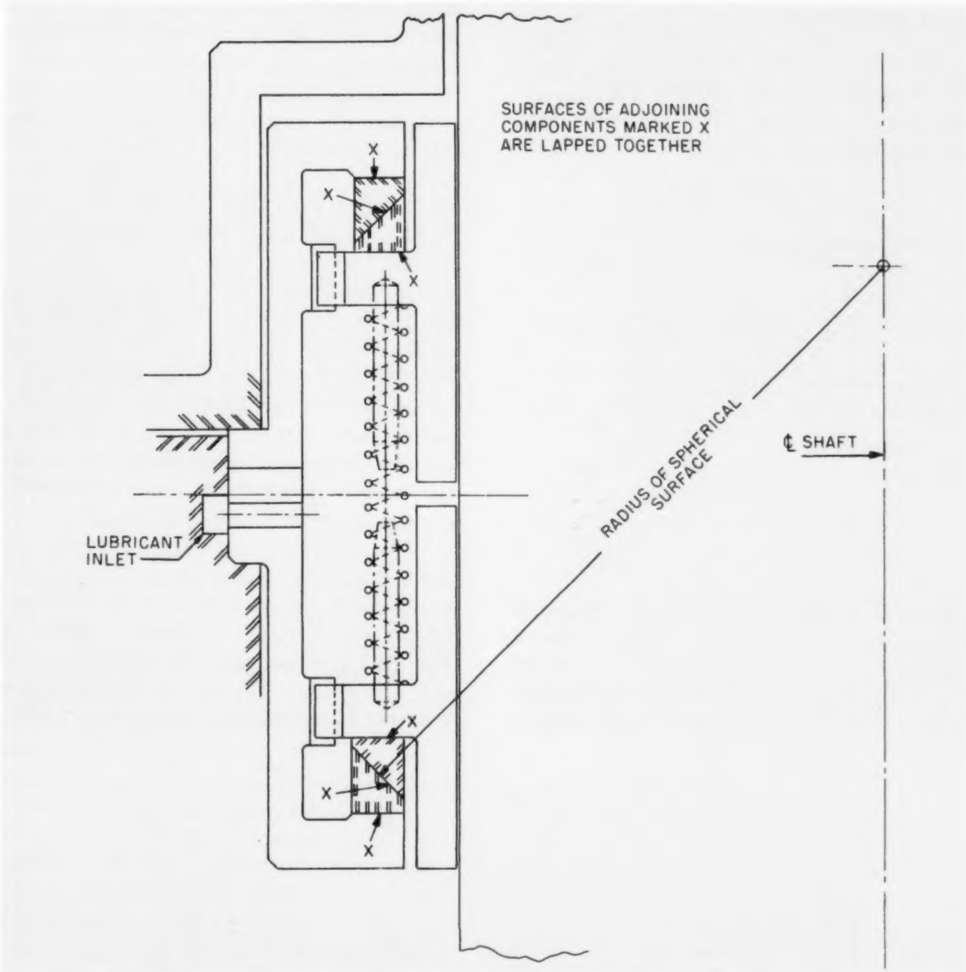

Fig. 15 Floating bushing seal with liquid lubricant.

rings. The cylindrical seal member can compensate for radial changes in shaft alignment by sliding on the faces of the spherical aligning rings and for angular misalignment by sliding on the spherical surfaces. Such a seal must be designed with careful consideration of temperature differences that could reduce the running clearance and result in seizures of the floating sleeves on the shaft. Such seals have been used extensively in small steam turbines, compressors, and centrifugal pumps. Economic studies should result in the selection of suitable materials and the most effective sleeve length to limit the lubricant flow to an acceptable volume.

Leakage-flow equations[13] applied to a floating sleeve seal (Fig. 15) with a 0.020-in. diametral clearance, a sleeve length ¾ of shaft diameter, and a lubricant pressure of 10 psig indicates an acceptably small leakage flow for the two bushings. The leakage flow varies:

- Directly as cube of radial clearance.
- Directly as pressure difference.
- Inversely as length.
- Inversely as absolute viscosity.
- Directly as shaft diameter.

The customary containment dam and slinger (not shown in Fig. 15) must be used to prevent the inward leakage of lubricant from reaching the sodium.

Centrifugal Liquid Seals

One type of centrifugal liquid seal (Fig. 16), in which liquid sodium is the sealing fluid,[10,14-16] employs a type of centrifugal impeller used in many steam-turbine designs. The centrifugal water impeller developed water leakage at the external atmospheric side, and both air and water leakage toward the inside of a steam-turbine casing when this seal was used at the exhaust end. The main difficulty is the formation of a mass of air bubbles in the atmospheric water annulus. When this type of seal was used at the vacuum end of a turbine, there was a constant boiling off from the internal water surface because of the heat generated by fluid friction.

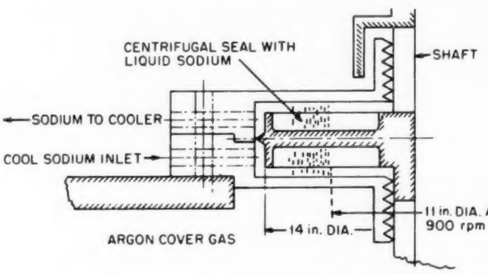

Fig. 16 Centrifugal liquid seal.

Figure 17 shows connections for removal and reentry of sodium at the impeller periphery for circulation through an external cooler. Because the liquid sodium must have a neutral cover-gas blanket, a liquid-lubricated upper seal is required (Fig. 12). The rotating impeller has radial vanes to ensure that most of the liquid metal is rotating at nearly the same angular velocity as the impeller. The low rotational speed of sodium pumps is an advantage for centrifugal seals. There is much less danger of forming a froth of sodium and inert-gas bubbles such as occurs in steam-turbine water glands, where the linear velocities are high. In the seal application shown, the tip velocity of the impeller is 55 ft/sec. In large steam turbines, impeller tip speeds exceed 200 ft/sec.

Operating centrifugal liquid-metal seals under shutdown conditions introduces complications. Using a centrifugal liquid seal in which MO-10 Fluorolube (see below) would be the sealing fluid appears possible. With such a design, a gastight seal is needed when the unit is shut down. This can be provided with an upper seal (Fig. 12). Centrifugal liquid seals require further

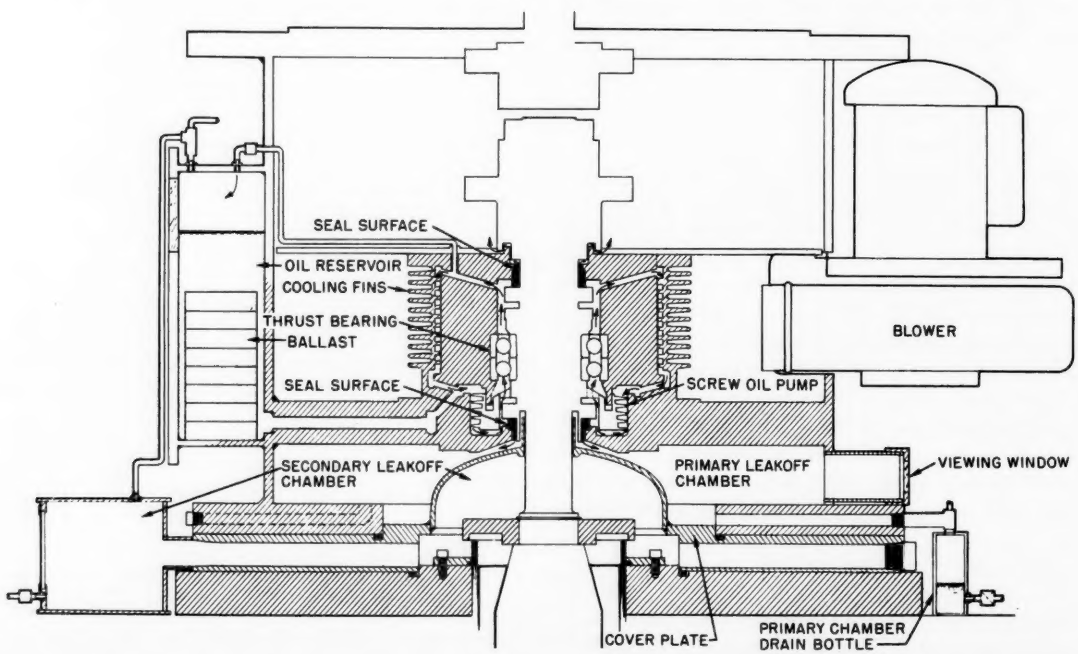

Fig. 17 SRE–PEP sodium-pump seal and bearing. (From USAEC Report NAA-SR-12409, 1967.)

design development before their practicability can be established.

SRE–PEP Upper Bearing and Seal Assembly

Upper bearing and seal assemblies (Fig. 17) for SRE–PEP sodium pumps[17] incorporate leakoff chambers that should be considered for future liquid-lubricant seals. Two independent chambers are incorporated in the design. The upper slinger, dam, and catchment basin have an independent drain tank, which is vented to the top of the lubricant tank. An independent lower slinger, dam, and catchment basin are provided to remove any lubricant that escapes the upper catchment chamber. The lower system has a separate drain tank.

Seal Lubrication

Long-term reliability needed with face-type seals requires positive lubrication of the matching faces. A standard type of turbine oil should be employed. Use of liquid lubricant to obtain supporting films of adequate thickness will result in considerable leakage to the argon cover-gas side of the seal assembly. This internal leakage must be removed to ensure against overflow into the sodium space. Design features to accomplish this are available.

The seals of the Fermi primary sodium pumps use MO-10 Fluorolube, a hydrogen-free lubricant. Produced by the Hooker Chemical Company, this lubricant currently costs approximately $191 per gallon. This lubricant was adopted because of the possible undesirable effects of hydrogen content in an accidental leakage of petroleum-base oil into the primary sodium space. Seal replacements during the several years of intermittent service of the Fermi reactor have not interfered with the operating program.

In general, gas-supported seals operate with very thin films unless excessive leakage can be tolerated. A flat-face ring is not ideal for hydrodynamic lubrication by either liquids or gases.

Gas-supported cylindrical bearings have operated satisfactorily with clean gas when the cylindrical surfaces of shaft and bearing are precisely lapped. Such bearings operate in a limited range of light transverse loads.

Seal Materials

Materials employed in ordinary bearing practice may be used for seal faces if there is a continuous supply of lubricant. Hard materials are desired because they have a better chance of survival if the lubricant supply fails. Chemically different materials of the maximum practicable hardness should be adopted for the face-seal elements. Materials selected should have high thermal conductivity and be noncorrosive. Materials that have proved effective in other applications are Stellite No. 6 and a hard tool steel. Reference 12, which describes Pratt & Whitney research, is an excellent reference on hydrodynamic face seals and materials.

Gastight Enclosure over Motor

An alternate sealing procedure is to enclose the motor and its bearings in a gastight casing (Fig. 18).

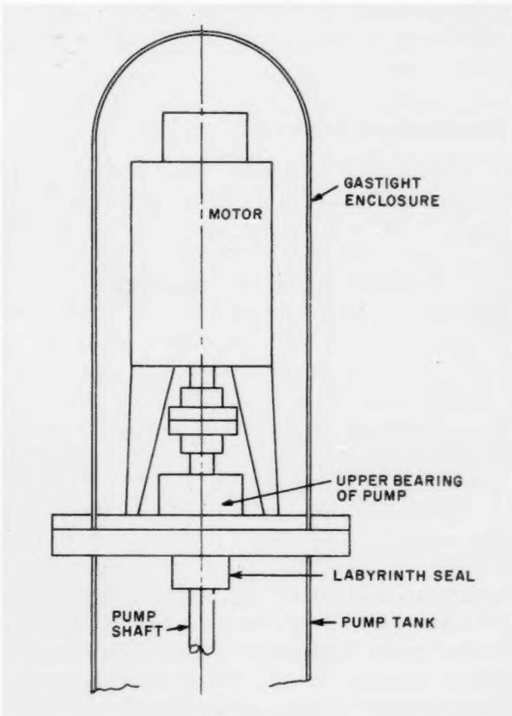

Fig. 18 Gastight motor enclosure.

This design simplifies the shaft-seal construction but introduces the requirement for a built-in cooling system for the motor. Factors affecting the design of gastight motor enclosures indicate:

1. There should be no electrical contacts of the switchgear or slip-ring type in the neutral-gas enclosure.

2. The motor bearings must operate in the neutral atmosphere.

3. The lubricants must not decompose to release hydrogen-containing gas or particulate matter to the cover gas and thence to the sodium space.

4. There should be a labyrinth gland long enough to minimize the diffusion of sodium vapor or particulate matter from the cover-gas space in the pump tank to the winding space of the motor.

5. A small continuous flow of argon gas should be admitted to the winding space of the motor to prevent the backward diffusion of sodium vapor.

Installation of Systems

Requirements for installation of pumps are due to such factors as forces on the pumps and the need for precise alignment.

Forces on Pump Structures

Sodium-pump components must be designed with careful consideration of the forces generated in each piping system and the associated stresses determined by calculation. Factors contributing to the accumulation of stresses include the transverse rigidity of large-diameter sodium piping connected to the pumps or pump tanks, the wide temperature ranges, and the use of austenitic piping materials with high thermal-expansion coefficients. The cold-springing offsets must be calculated and followed by the appropriate erection procedures for the three translations and three rotations at each final makeup joint.

Procedures established by A. M. Wahl, A. S. McCormick, C. T. Mitchell, J. H. Walker and others for dealing with expansion forces in complex piping systems are well known and accepted. An excellent reference is the *Piping Handbook* by R. C. King and Sabin Crocker (McGraw-Hill, 1967). *The American Standard Code for Pressure Piping,* supplemented by a section on Power Piping, includes valuable data. Failure to establish the maximum permissible bending and torsion moments for each connecting pipe can lead to serious trouble. Builders of centrifugal pumps and pump tanks must specify the maximum translation forces and moments that may be safely imposed on the respective pipe flanges and other mechanical connections. The piping designer and contractor must ensure that the expansion forces of the piping system do not exceed the limits established by the designers of the sodium-pump components.

General Arrangement

Experience with sodium pumps in service and installation and operating problems indicates the following design features are desirable: (See Fig. 19.)

1. Vertical pump shaft with direct-connected motor.

2. Two-bearing pump shaft with lower bearing of the hydrostatic or "fluid-piston" type, submerged in the sodium. Supporting areas of the lower bearing to be supplied with sodium from the pump discharge.

3. Upper pump-shaft radial bearing located above the seal in an easily accessible position and lubricated with a standard turbine oil. A cooling system is required for upper-bearing oil.

4. A thrust bearing of the Kingsbury tilting-pad type for both pump and motor, located in the upper radial-bearing housing.

5. Bolted-flange coupling above upper-bearing housing to connect motor and pump shafts, with provision for checking alignment. This operation will require removing coupling bolts and separating shafts to remove transverse and angular restraints. Adequate space must be provided for checking alignment without removing motor.

6. Motor support designed with means for transverse and angular adjustments to secure coincidence of motor- and pump-shaft axes.

7. Providing for the removal of a pump from the system in every sodium-pumping installation. Apparatus is necessary for connecting a collapsible plastic bag to a flange in the floor, after which the pump assembly can be raised vertically into the distended plastic bag. Special sliding gates are then closed to maintain an argon blanket in the bag and in the pump tank. The pump assembly, carrying an amount of adhering liquid sodium, can then be lowered into a containment tank or removed from the plant after suitable safety precautions have been observed.

8. Standards of reliability for shaft-seal members. Seals must be replaced in the minimum possible time. Designs must be developed so that the lower coupling half can be removed without removing the motor assembly. Doing so will require a method for sealing the cover-gas space against objectionable leakage when a seal is being dismantled for inspection or replacement. The design should ensure that the seal rings can be removed and replacements installed when the lower coupling half is removed. The lower coupling half can be of the "muff" type, split axially.[18]

9. Since the distance between bearings of some of the pumps can be expected to exceed 20 ft, critical

speed of the vertical shaft will probably require a hollow shaft, such as used at EBR-II and Rapsodie.

10. Approximately located and spaced radial baffles to prevent vortex formations in the pump tank.

11. Except for bearings, 304 stainless steel[19] is an accepted material for the main pump components that are exposed to sodium or sodium vapor.

12. Design of means for retaining petroleum-base oils for the upper bearing and for the motor to ensure that oil vapor or direct leakage will not enter the sodium containment.

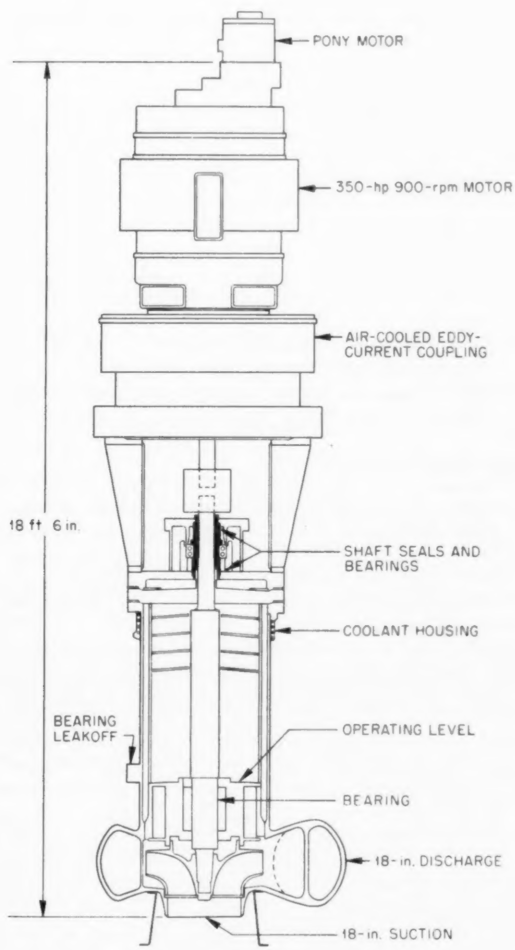

Fig. 19 Fermi secondary sodium pump. (From page 179 of Ref. 4; this figure appeared originally in Enrico Fermi Atomic Power Plant Technical Information and Hazards Summary Report, 1962.)

Alignment

Adequate provisions must be made to facilitate the precise determination of the alignment of the motor and pump shafts with the motor in place. The necessary measurements require the motor and pump shafts to be joined by a bolted coupling, which can be separated for axial measurements between the faces. Measurements between coupling faces at the extremities of two diameters at right angles, followed by a second set of measurements at the same diameters when both shafts are rotated 180°, give the necessary data for determining parallelism of the axes of the two shafts.

A series of readings with dial indicators mounted 180° apart on one coupling flange, arranged to indicate the transverse position of the outer circumference of the opposite coupling flange, will determine transverse location. It is important to provide sufficient space for taking the necessary measurements.

The designer must provide means for changing the angle of the motor axis. One method uses two annular support rings with the junction at a small angle so that the top and bottom surfaces are parallel when placed together. By rotating one ring relative to the other, the angle of the motor axis can be corrected.

The preferable method of doweling uses four radial cylindrical keys, 90° apart, between the bottom surface of the support structure and the sealing plate. The holes for these keys should be drilled and reamed after the angular and transverse alignment corrections have been made.

Large sodium pipes will introduce large thermal-expansion forces on the pump tanks during certain phases of the operating cycle. The alignment of pumps and motors of primary and secondary sodium pumps should be checked when the system is at minimum and maximum operating temperatures.

Conclusions

Technical papers on shaft-seal design indicate a widespread interest in this development and an increase in the understanding of the seal problem. Emphasis has been placed on the face-type seal. Future development will emphasize rigidly mounted rotating collars with stationary seal collars of minimum practicable mass to permit their following the residual errors in truth of the rotating faces. Equally important is the development of a key system for the stationary rings that will permit unimpeded angular alignment.

The gas-supported seal offers opportunity for development. Problems to be solved include loss of argon cover gas on the atmospheric side and providing positive means to separate the metallic seal faces by either hydrodynamic gas films or pressure-support areas. A centrifugal liquid seal in which a compatible oil is employed might be applicable.

The sealed enclosure over the motor remains attractive because the construction is relatively simple and trouble-free. Problems include supplying the winding space of the motor with clean argon gas and developing a labyrinth shaft seal to permit a small flow of argon gas to the sodium cover space at a velocity exceeding the back diffusion velocity of the sodium vapor. The economic evaluation of this arrangement must include the closed cooling system for the motor.

The alignment and the effects of thermal-expansion forces on pump structures are most important in planning future plants.

Electromagnetic pumps do not appear applicable to the large power plants of the future.

Acknowledgments

The author wishes to express his appreciation to the Argonne National Laboratory, Atomic Power Development Associates, Inc., the Power Reactor Development Company, and the Allis-Chalmers Mfg. Co. for assistance in the preparation of this article. Figures 1 to 7, 11, and 19 of this article are reproduced from the book *Fast Reactor Technology: Plant Design*, John G. Yevick and A. Amorosi (Eds.), The M.I.T. Press, 1966.

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Sodium-Cooled Reactors: Highlights of Operating Experience in Plant Systems

By Myrna L. Steele*

Events and problems related to the operation or performance of sodium-cooled-reactor plant systems during the approximate period January 1966–January 1969 are summarized in this article to provide an information base that can be updated periodically. In this way we can keep the reader aware, on a continuing basis, of occurrences and pertinent documents.

It will be readily evident that no information is included below on physics aspects, instrumentation, or management, e.g., training, missions, and achievement. Operating experience with plant systems, only, is summarized for EBR-II, the Fermi Fast Breeder, the Dounreay Fast Reactor (British), Rapsodie (French), and the BR-5 (Russian). In the next issue (Fall 1969), we plan to summarize the physics aspects of these reactors and review any material published on plant systems since January 1969. The Winter 1969–1970 issue will have a summary of material on instrumentation, with particular emphasis on in-core instrumentation, and a review of any new information on the plant-systems and physics aspects. The Spring 1970 issue will include a summary of information on the management of these reactors and will update any previous material. From then on we plan to keep the reviews as current as possible.

The reader should bear in mind that, generally, these summaries of occurrences will be failure-oriented, i.e., attention will be centered around component or systems malfunction or malperformance. Problems caused by these failures will be described, and, where possible, failure causes and solutions to problems will be indicated. Extensive, although possibly not exhaustive, references and reading lists are included.

Experimental Breeder Reactor II (EBR-II)

Between January 1966 and January 1969, the 62.5-Mw(t) EBR-II logged 17,195 Mwd of successful operation.[1,2] Thermal and neutronic performance has been essentially as predicted with the possible exception of the so-called reactivity hysteresis,[3] or reactivity draft, which will be discussed in detail in a future issue of *Reactor and Fuel-Processing Technology*. The reactor and primary system have been highly stable and readily controllable; mechanical stability of core components has been good. Fuel-element examination after the first core was unloaded gave no indication of wear or vibration; no fretting marks were observed. Maintenance and repair or replacement of components have been accomplished.[4]

Reactor Oscillator. The oscillator, which was installed for transfer-function measurements, has given trouble. In February 1967 it was noted that the sensing rod on the oscillator drive shaft was binding and that manual assistance was required to move it. The presence of some foreign material or possible galling between the sensing shaft and the jaw-actuating sleeve was indicated by the fact that the sensing rod moved with the jaw-actuating sleeve; these symptoms could also indicate a broken sensing-rod bellows.[5] Subsequent disassembly and inspection, by March 1967, revealed a bellows rupture that had caused sodium oxide accumulation, and, in turn, sensing-rod binding.[6]

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Primary and Secondary Pumps. Both 5000 gal/min mechanical centrifugal pumps[7] (see Fig. 1 of Robert C. Allen's article in this issue) failed during preoperational testing because the pump shafts rubbed at the lower labyrinth seals. The pumps were reworked, and an argon backflow was added to prevent the shafts from sticking because of sodium oxide accumulation.[4] In February 1968, problems with the controls for the primary pumps were reported;[8] by September 1968 an average of three minor perturbations of 1 to 4% flow at random intervals was reported for each day during reactor runs. Analysis showed that some of the problems were due to breakdown of the thyratron control tubes after about 2000 hr of operation. These tubes are now being studied to determine their oscillation behavior.[9]

Flow-coastdown tests indicate that the secondary sodium a-c electromagnetic linear induction pumps will function at least ½ hr with low flow, reverse head after cooling-water loss.[1]

Primary and Secondary Plugging Meters. In March 1967 a flow decrease in the primary-system plugging-meter loop was noticed; attempts to restore normal flow revealed copper deposits on the plugging-meter valves. Upon inspection the electrodes of one of the auxiliary primary pumps revealed evidence of severe pitting. Water-displacement measurements indicated that approximately 10 lb of copper was missing from the two electrodes. The electrodes were repaired and modified by installing a stainless-steel sleeve over the copper to provide a continuous covering over the conductors and electrodes.[6] In February 1968 a fire in the sodium boiler plant destroyed many components of the secondary sodium plugging loop. The components have been replaced.[8]

The heaters for the primary system had to be replaced during the January 1966 semiannual maintenance; heaters of a different design were installed, and these have performed satisfactorily.[1]

Fuel-Handling System. Fuel-element insertion and removal from the core are accomplished in the EBR-II by two concentric rotating-shield plugs that have a fixed fuel-handling machine attached (Fig. 1). A Bi–Sn eutectic is used as a seal between the plugs and is contained in a trough between the large and small plugs. During initial testing and operations, difficulties were encountered in turning the large rotating plug to position the fuel gripper.[8] Problems were traced to accumulations of oxide deposits in the outer annuli of the trough. Vacuum cleaning was tried with poor results; however, coring and a wire-brush cleaning technique resulted in the removal of about 300 lb of alloy and oxidized material from both troughs. The coring and brushing are now used on a periodic basis, and plug rotation has remained smooth.[1]

Accumulation of sodium or sodium oxide on the fuel-unloading-machine gripper mechanism caused the gripper to stick and necessitated removal for cleaning.[8] Difficulties with the main core gripper were reported in July 1967. Problems in releasing an engaged fuel element were encountered, and loss of rotational control of the subassembly was evident.[10] By August 1967 the core gripper was replaced by a spare unit, and normal fuel-handling operations were resumed. The reason postulated for the gripper problems was physical damage to the gripper; however, disassembly and remote inspection revealed none. The gripper was transferred to the hot laboratories for detailed inspection.[11]

Steam-Generating System. Loss-of-load tests have been accomplished without interruption of reactor operation; separation of load demand has been handled smoothly with only small transients indicated by steam-system and reactor-control instrumentation.[1]

During a scheduled maintenance shutdown in September 1966, the main turbine was disassembled for a routine 10,000-hr-operation inspection. Damaged blading was found in the second and third stages with slight damage as far down as the seventh stage. The damage was apparently caused by a portion of a spill strip which broke from the turbine casing in the area of the first stage. The manufacturer's representative repaired or replaced all damaged parts.[12] In-place repair of tube-to-tube-sheet welds in the steam generator (Fig. 2) was accomplished during initial testing; only minor maintenance has been required during routine operations.[1]

In September 1967 the high-pressure steam valves and steam-drum safety relief valves required repacking, lapping, and resetting.[13] Misalignment of the turbine exhaust nozzle caused loss of inboard pump bearings in the turbine-driven condensate pump[3] in December 1967.

Generally the instrumentation has proved satisfactory. Problems with the primary system occurred during preoperational and approach-to-power tests; 9 of 12 resistance thermometers, 13 of 17 pressure sensors, and 2 of 5 modified venturi-type flowmeters failed.[4] The nuclear instrumentation performed quite well. Details on instrumentation, with emphasis on in-core instrumentation, will be covered in the Winter 1969–1970 issue of this journal.

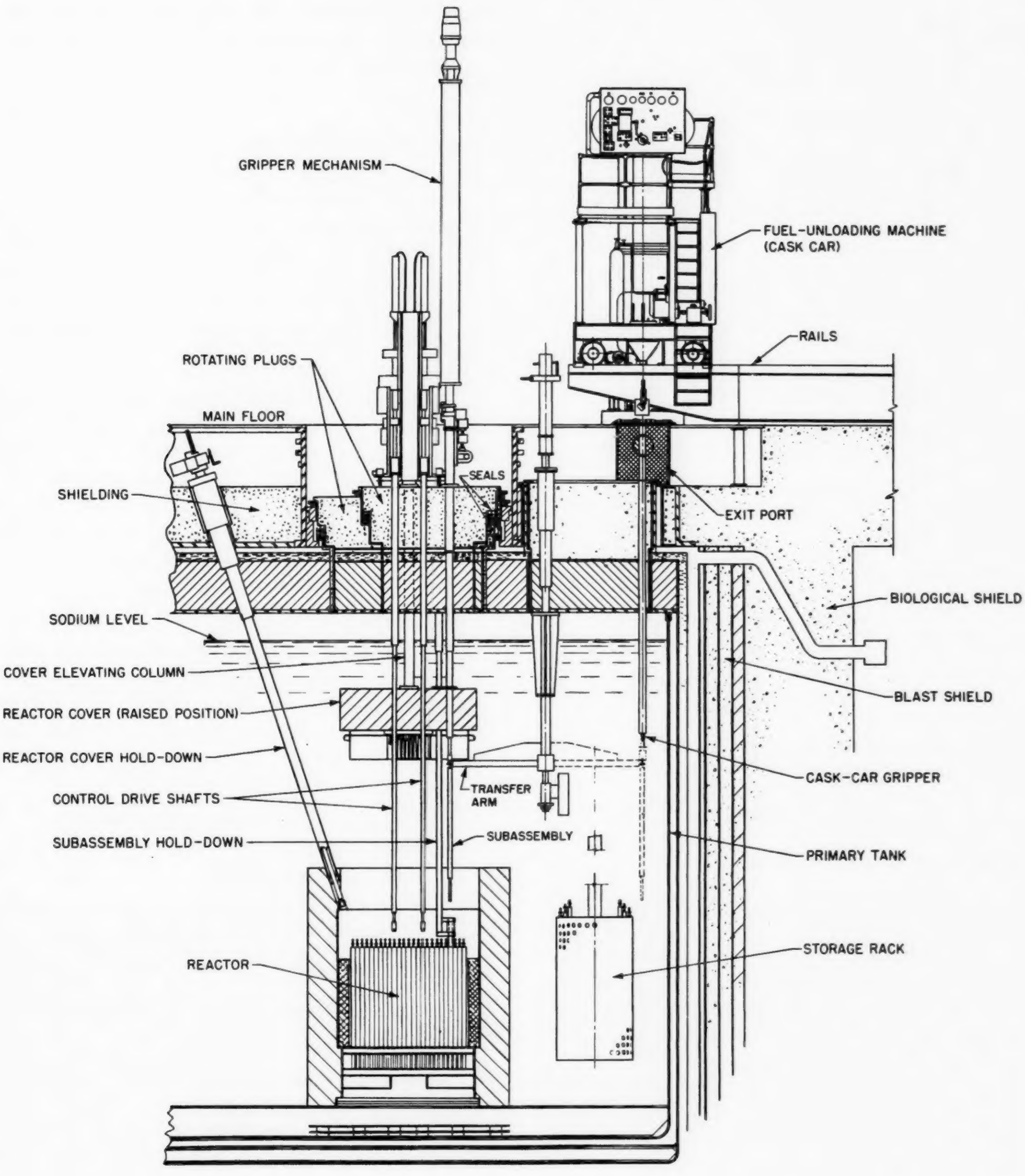

Fig. 1 EBR-II principal fuel-handling components. (From page 383 of Ref. 7; this figure appeared originally in USAEC Report ANL-5719.)

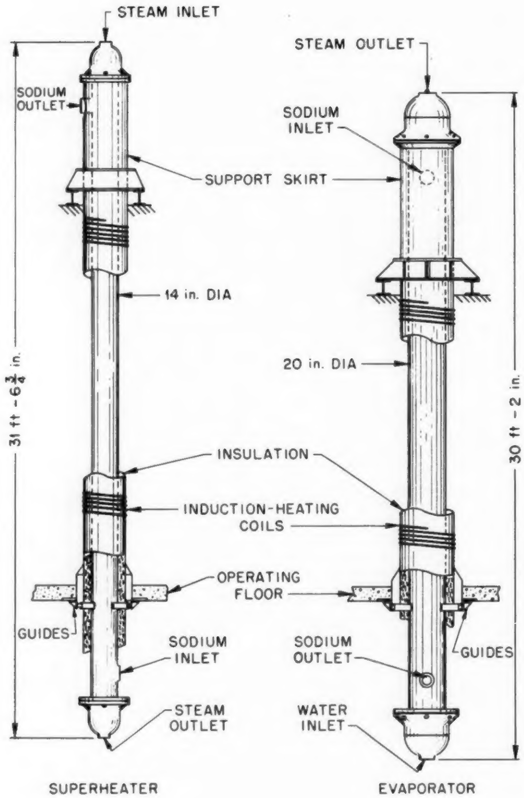

Fig. 2 EBR-II steam generator. (From page 215 of Ref. 7; this figure appeared originally in USAEC Report ANL-5719.)

Fermi Fast Breeder Reactor

Before its provisional operating license was issued, the 200-Mw(t) Fermi Fast Breeder Reactor was used for development of components and systems for a sodium facility. As a consequence, much information had been obtained on the functioning and performance of these reactor components and systems in a sodium environment before the postneutron phase of tests. Also, procedures and techniques had been developed[14] for sodium-environment maintenance work on practically all components or systems associated with sodium-cooled power reactors.

Primary and Secondary Pumps. The 12,000 gal/min mechanical shaft-seal centrifugal pumps for primary sodium (see Fig. 2 of Robert C. Allen's article in this issue) have logged about 30,000 hr of successful operation in sodium at speeds of 300 to 860 rpm and temperatures of 450 to 1000°F. Hydrostatic bearings and pump parts immersed in sodium have required no maintenance; however, the oil-lubricated inert-gas seal face (Fig. 3) has required frequent replacement. Improved seals are being designed to correct the problem. The secondary sodium pumps have logged about 18,000 hr of successful operation; these pumps are very similar to the primary pumps except that they have a shorter shaft.[15]

Primary sodium-overflow pumps were originally equipped with both labyrinth and face seals. The labyrinth seal was removed when it became bound and caused a pump to seize. The face seal has since required frequent replacement. No problems have been encountered with the five sodium-lubricated sleeve bearings.[15]

Primary-System Check Valves. The original primary-system check valves were the swing-disk type with a downflow configuration. This type of valve was satisfactory with respect to pressure drop, but the initiation of reverse flow resulted in strong sodium hammers that caused considerable pipe movement. The valves were replaced with a spring-loaded disk type with a dashpot to reduce closure force.[15]

Heat Exchanger. Mechanical performance of the intermediate heat exchanger (Fig. 4) has been good, but, on the basis of a small amount of data at low loads, the thermal performance appears to be about 50% of the manufacturer's rating.[15,16] Possible reasons for the poor performance are probably a combination of:[17]

1. Shell-and-tube-side flow maldistribution.
2. Ineffective heat-transfer area.
3. Shell-side flow stagnation.
4. Shell-and-tube-side fouling.
5. Shell-and-tube-side flow bypassing.
6. Reduced heat-transfer area caused by tube sleeves.
7. Cover-gas entrainment on the heat-transfer surfaces.

Flowmeters. The electromagnetic flowmeters for the primary and secondary sodium systems were calibrated at low power and assumed to be linear; however, pump tests at higher reactor power levels showed them to be nonlinear and temperature sensitive. At operating flow levels, they were in error by about 30% because the error increased with higher flow rates. The error was primarily attributed to distortion of the magnetic field by eddy currents in the sodium.[16]

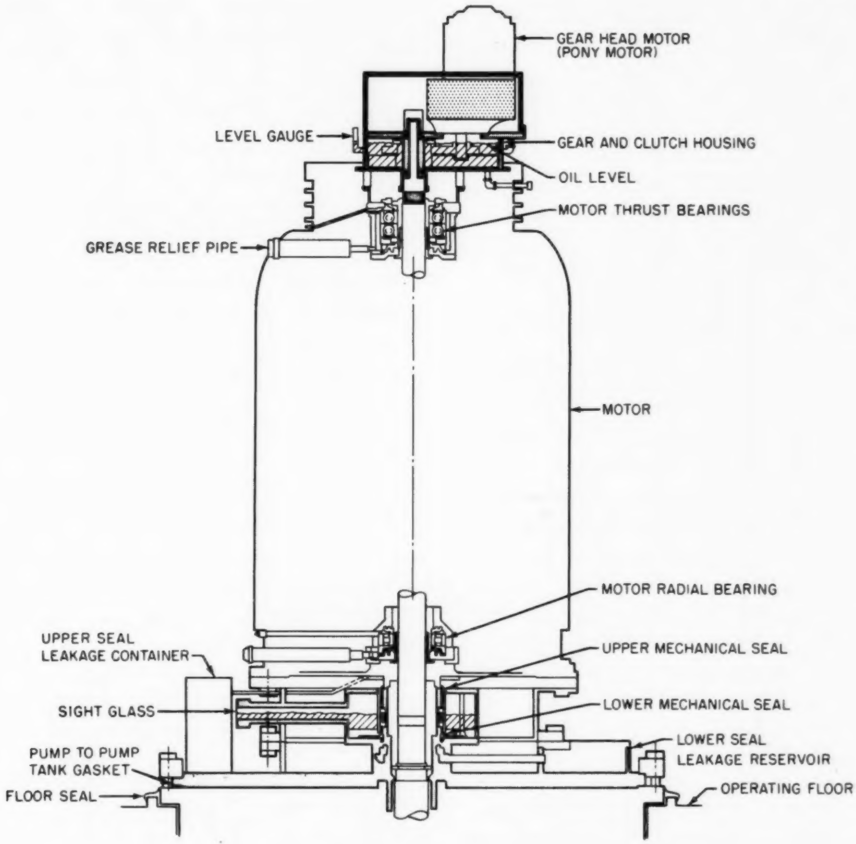

Fig. 3 Fermi primary-pump shaft-seal details. (From page 177 of Ref. 7; this figure appeared originally in the Enrico Fermi Atomic Power Plant Technical Information and Hazards Summary Report, 1962.)

Steam Generators. In 1962 a sodium–water reaction occurred in the No. 1 steam generator; the leak detection, reaction relief, and isolation and dump systems contained and terminated the reaction with no external effects. Tube vibration had caused a leak in one of the welds, allowing a water-to-sodium leak; impingement baffles and tube lacing clips were installed to reduce vibration. Figure 5 shows the final steam-generator design. Leaks in tube-to-tube-sheet welds have occurred almost continuously over the 7 years of sodium service. During the plant downtime after the fuel-meltdown incident (discussed later), a program was initiated to repair all 1200 water-manifold tube-to-tube-sheet welds in each of the three steam generators. The method used for weld repair was a tungsten–inert-gas method which uses a special welding head inserted into a steam-generator tube to fusion-weld the tube to the tube sheet.[1]

During low-power operation of the steam generators, high steam-generator sodium-outlet temperatures initiated three single-circuit shutdowns which, in turn, initiated unscheduled scrams. These instabilities were caused, in part, by boiling in the downcomer tubes; the feedwater reheat area was changed from 1500 sq ft to 1700 sq ft, and most of the instabilities disappeared.[15,16]

Feedwater Flow-Control System. Resistance-thermometer detectors (RTD's) were originally used for the input signal to the automatic flow-control system for the steam generators. However, the slightest boiling in the steam generator produced oscillations, which, in

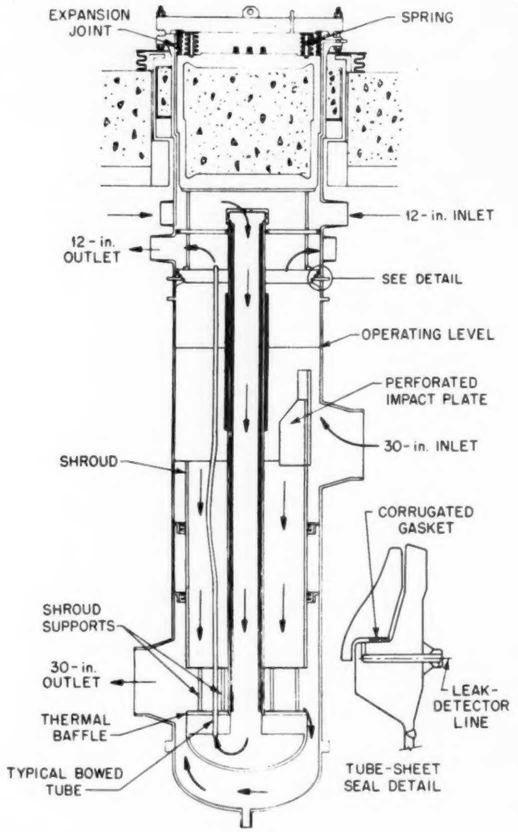

Fig. 4 Fermi intermediate heat exchanger. (From page 196 of Ref. 7; this figure appeared originally in the Enrico Fermi Atomic Power Plant Technical Information and Hazards Summary Report, 1962.)

turn, produced an erratic response in the RTD's. The RTD's had to be replaced with rapid-response thermocouples. Further difficulties were encountered in sequencing the three valves that are in parallel in the feedwater flow-control system; these difficulties were overcome by trial-and-error timing.[16]

Fuel-Handling Systems. The cask car and fuel-transport facility have given constant problems having to do with the argon atmosphere, design deficiencies, decontamination, and maintenance.[16]

Emergency Power Supply. The emergency power-supply system has been called upon to function once. Two cable faults caused loss of off-site power. The diesel generator started automatically and maintained emergency power until the off-site supply was restored.[18]

Fuel-Meltdown Incident. In September 1966, three unscheduled scrams were caused by single rod drops. The rod magnet decay current had a peculiar characteristic; latching problems were also experienced with one rod. This control rod was removed for latch-magnet and gripper-head inspection. In addition, three thermocouples that indicated coolant exit temperatures from the fuel assemblies had registered higher than normal. The fuel assemblies associated with the three high-reading thermocouples were relocated to see if the high readings were position dependent or peculiar to the fuel assembly itself. The only change noticed was a loss of 3 cents in reactivity between the first and second rod stops; this reactivity loss was attributed to the control-rod removal.[19]

On Oct. 5, 1966, at 0345 the Fermi Fast Breeder Reactor was at 1 Mw(t); there was good correlation between rod position and previous reactivity measurements relative to sodium temperature. Power-level

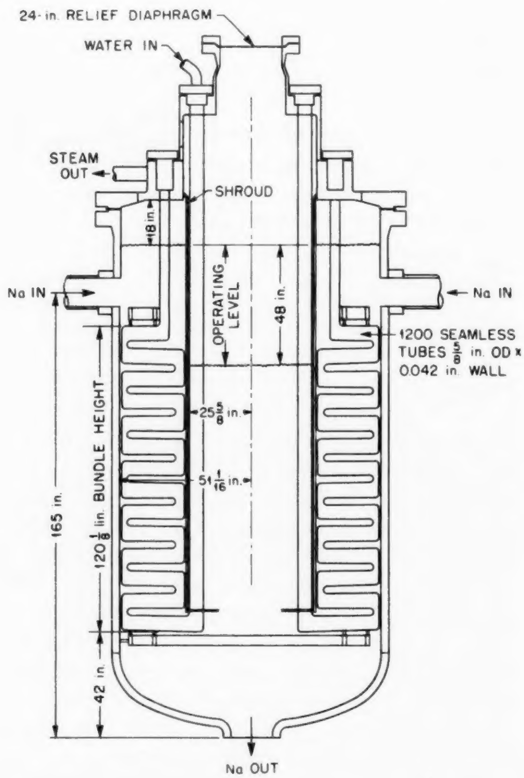

Fig. 5 Fermi steam generator. (From page 212 of Ref. 7; this figure appeared originally in Report APDA-124, January 1959.)

increase was begun at 1345; at 1500 the operator noticed an erratic *dn/dt* signal. The reactor was switched from automatic to manual; the *dn/dt* signal settled out. The erratic signal was considered to be electronic noise because no instability or nuclear transient was indicated and because similar response to electronic noise had been experienced in the past. At 1505 the feedwater control was put on automatic, and once again the *dn/dt* signal became erratic. A staff engineer noted that the control-rod positions were higher than previously registered for this power level. A check of core outlet temperatures revealed that two thermocouples were indicating higher than expected for a nominal 34-Mw(t) reactor power level. At 1509 the high-radiation alarm signals registered in the control room from the reactor-building monitors. The building was isolated and restricted; no one was in the building at the time of the alarms. The monitor in the fission-product-detection building exceeded its set point; automatic isolation of the fission-product-detection system followed immediately. Power was reduced manually; still the thermocouple readings remained high. At 1520 the reactor was manually scrammed from 29 Mw(t). The micromicroammeter indicated that all six safety rods dropped.[20,21] A high fission-product release to the argon cover gas and a loss of 22 cents in reactivity were noted. No appreciable increase in sodium-flow resistance through the core was observed. Later, tests showed the 22-cent reactivity loss to be permanent; pump flow tests and flow coastdown indicated a 2 to 4% reduction in flow, which is on the order of measurement accuracy.[20] By January 1967 it had been determined that at least two fuel assemblies were either jammed or melted together;[22,23] special tools were obtained, and the assemblies were remotely inspected.[24]

In April 1967, during unloading of the undamaged fuel from the core, the exit-port plug seal on the offset handling mechanism allowed some cover gas to leak out; the seal was replaced. Then, after the shipping pot had been lowered onto the transfer rotor, the pot gripper would not release, and the segmented gripper separated under 10 ft of sodium. The tool was recovered, and the gripper and shipping pot were removed from the reactor. The Inconel 600 gripper stem was replaced by a Stellite 6B stem; more modifications are to be made to this tool.[25]

The sodium was drained and the damaged fuel inspected[26] in June 1967. The fuel assemblies were separated by a specially fabricated tool and removed in July. In the process of loading the damaged fuel into a transfer pot, the lifting cable broke; a subassembly, transfer pot, gripper, and one extension tube dropped 20 ft onto the transfer rotor plate. The rotor plate was dented slightly but still functioned. And then, during an attempt to sample some of the material deposited on the fuel-assembly separation tool, a leak developed in the gas seal around the shipping cask causing contamination of the containment building. Even after the removal of the damaged fuel assemblies, there was still no clue as to what had caused the meltdown.[25]

Storage of the primary sodium was completed in August 1967. Safety-rod guide tubes, wrapper cans, support plate, and subassembly seats were inspected for mechanical damage; none was found.[26] In September an unidentified foreign object (UFO) was sighted in the meltdown pan.[27] Several different tools were designed and fabricated, and attempts were made to retrieve the UFO; nothing seemed to work.[27-30] Fabrication of a mockup of the high-pressure plenum with the 14-in. inlet was begun in October.[28] In January 1968 the UFO was identified as a segment of the zirconium conical flow guide which had been installed in the lower plenum. Efforts toward object removal were then directed to the 14-in.-line.[31]

In June 1969 four pieces of the zirconium flow guide had been removed from the primary system, and plans were in the final stages for the reloading and subsequent operation.

The following recommendations and suggestions by Atomic Power Development Associates and Power Reactor Development Company personnel are direct products of the evolutionary experience gained from present-day fast breeder reactors:[32]

1. Flowmeters should be calibrated before, not after, installation, and the system should facilitate recalibration at any subsequent time.
2. The sodium systems should have provisions for trapping and removing foreign particles; the Fermi design is not entirely satisfactory in this respect.
3. Thermocouples and resistance heaters should be particularly accessible because they have large failure rates; in Fermi these components are essentially unreplaceable after the systems become radioactive.

Although the last 3 years have been replete with problems for those who own and operate Fermi, much invaluable experience has been accumulated on operation and behavior, under both normal and abnormal conditions, for the fast reactor program as a whole. Much information resulting from this experience has been documented. The information covers maintainability, repairability, removability, and access, as well as the relation of these to overall reactor design.

Programs for the dissemination of this information will be discussed in a future issue.

Dounreay Fast Reactor (DFR)

The 72-Mw(t) DFR, at Caithness, Scotland, has 24 coolant loops, each of which is provided with a pump and a heat exchanger. In 4 years at full power with 39,000 Mwd of operation, the behavior of the DFR has been largely predictable.[33] In the early days, oxide impurities in the liquid-metal coolant caused a control-rod problem. This problem was solved, and no further control-rod problems have occurred. However, the control system itself has been simplified. The number of control rods was reduced from 12 to 8, and insertion speed was increased by a factor of 10. In addition, the control-rod drives were changed from two-speed to uniform-speed.

Heat Exchangers. An intermediate heat exchanger (IHX) (Fig. 6) developed a leak after 8 years of being filled with liquid sodium. The leaking sodium caused

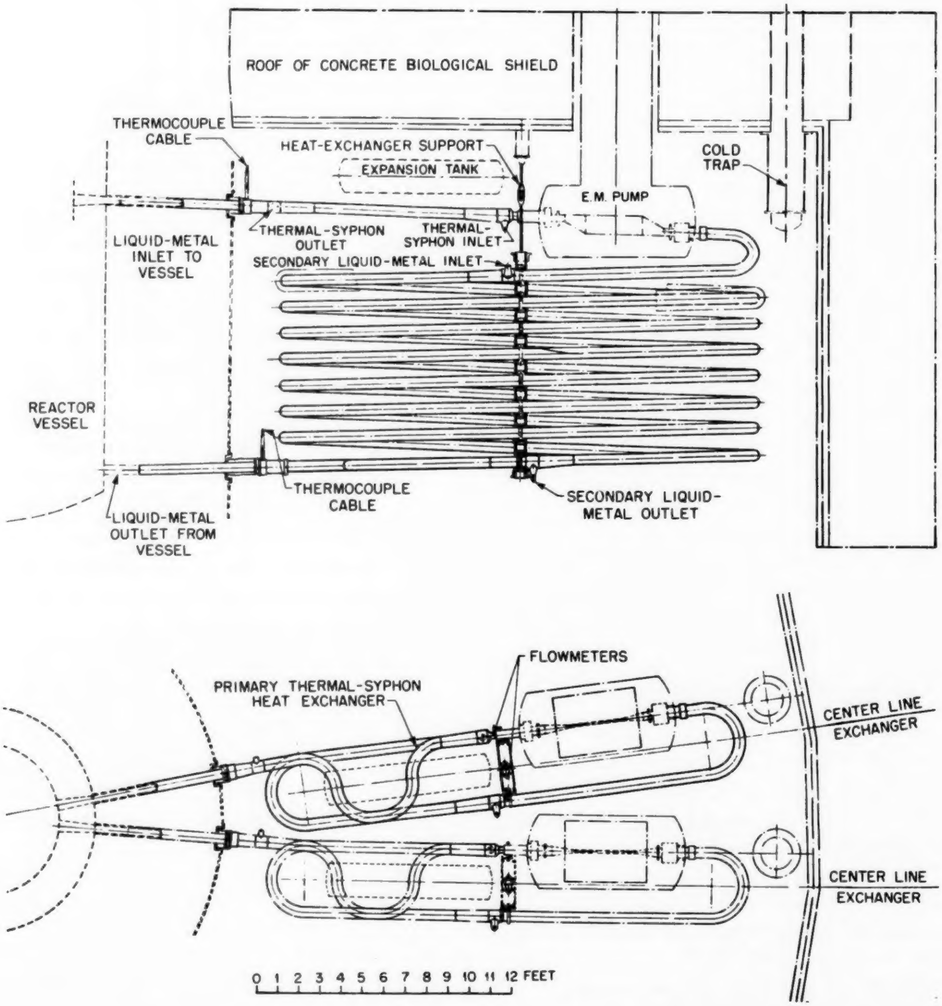

Fig. 6 DFR intermediate heat exchanger. (From page 199 of Ref. 7; this figure appeared originally in R. R. Matthews et al., Proceedings of the Symposium on the Dounreay Fast Reactor, London, Dec. 7, 1960, Institution of Mechanical Engineers, London, 1961.)

corrosion of the areas it touched, and the IHX was out of service for 10 days. Because the leak was small, it had gone unnoticed or undetected for some time. The incident showed the limited sensitivity of the leak detectors for small leaks. As a result, more sensitive detection methods are being developed.

Primary System. On May 9, 1967, a leak detector indicated that primary coolant was escaping, and the reactor was shut down. After a few weeks of being unable to find the leak, workers installed a better leak detector and started up the reactor. The new leak detector showed that about 1 liter of coolant was lost every 3 to 4 min. On July 29 the reactor was shut down for further tests because the leak rate was increasing slowly. However, each time the reactor was shut down the leak stopped; therefore the leak appeared to be coolant-flow or pressure induced.[33] In late 1967 the leak was located in one of the 24 heat-exchanger-connecting pipes about a foot from the bottom, in the area of the reactor-vessel nozzle. The leak has been repaired, and the DFR is back on-line.[34]

Rapsodie

The 20-Mw(t) reactor Rapsodie, located at the Cadarache Research Center, France, achieved criticality in 1967. Some significant experience had been accumulated prior to loading of the core.

Sodium Spillages. During preneutron testing of Rapsodie in October 1966, two incidents of sodium coolant spillage occurred. The first spillage occurred during preparations for filling the pressure vessel when the secondary lines were being preheated. The sodium had previously been allowed to freeze in the lines; during the preheating, two spots in the piping proved harder to heat than the rest. As a result, when the sodium between these two spots did melt, it had no expansion room, and consequently the piping burst on the containment-building side of the piping. This area of piping is double walled, and so the pipe rupture passed unnoticed. Therefore, when the filling with sodium began, the whole double-walled loop and the internal structure of the IHX (Fig. 7) became filled with sodium. The IHX had to be completely dismantled and cleaned. During this time the improperly constructed leak-detector system was replaced with an improved one, containment penetrations were modified to allow improved preheating, the piping in the double-walled area was segmented to preclude filling of the entire loop if future leaks or ruptures should occur, an emergency drainage pipe was added to the secondary loop to limit the amount of coolant leak, and thermocouples were added along the system to provide temperature readout.[35]

The second spillage occurred in about the same period as the first and was caused by improper opening of the gate valve between the argon cover gas and the primary sodium. An overpressure of argon in the primary loop caused the rotating plug to be filled with sodium. While the sodium oxide was being cleaned from everything, strong-flow mercury valves were substituted for the small-flow mechanical safety valves, the excessively prompt electric control on the equilibrating valve was reduced, a direct decompression pipe was installed from the storage tank, and improved pressure-surveillance instrumentation was installed in the primary system.[35]

Pump Seizures. Two pump seizures occurred during preoperational testing. First, a secondary sodium pump seized. The cause of the seizure was not determined, but, since the seizure had occurred gradually, vibrations in the framework or piping were suspected. Connections between the inlet piping and the shell were reinforced, and surveillance instrumentation was added. Second, a primary pump (see Fig. 3 of Robert C. Allen's article in this issue) seized under much the same conditions. The materials for the hydrostatic bearings were checked, and new bearings of another material were installed.[34]

BR-5 Reactor

As of July 1, 1967, the 5-kw(e) BR-5 reactor at Obninsk, USSR, had produced 9.86×10^4 Mw-hr of electric power.[36]

Primary System. Cold traps keep the oxides down to less than 3×10^{-3} % to prevent the corrosion of austenitic steels and the plugging of small orifices throughout the system.

Four valves in this system have had to be replaced because of bellows leakage. The leakage, each time, was duly registered by the leak detectors. There has been no sodium leakage through the gland packing.

Problems with the coolant-level indicators leaking sodium have been encountered five times in the 7 years of operation. Local combustion was observed as a red glow in the area of the level-meter flange in both the primary and secondary systems. The leaks occurred in welds between sections of the measuring tube of the level indicator.

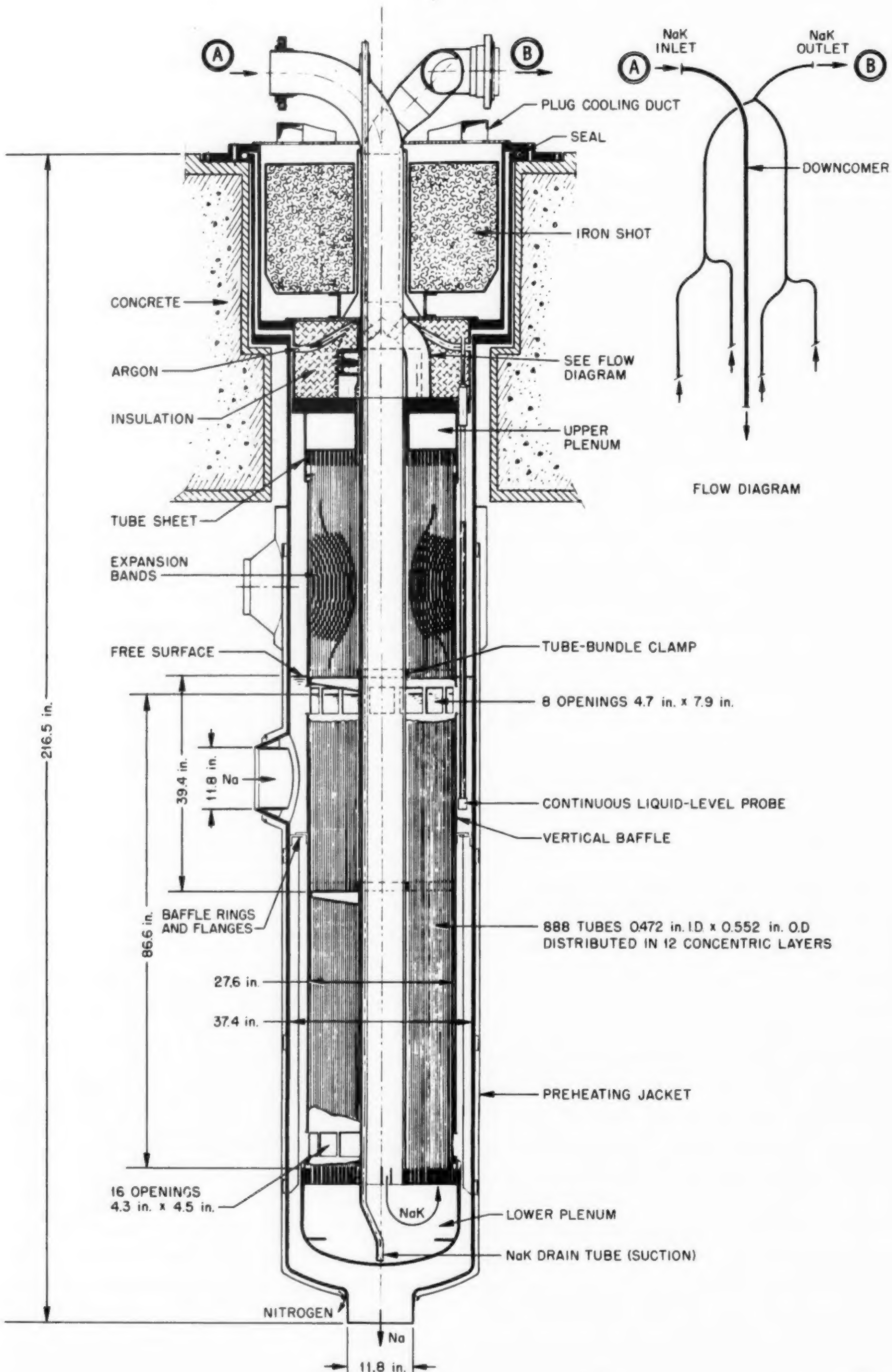

Fig. 7 Rapsodie intermediate heat exchanger. (From page 200 of Ref. 7; this figure appeared originally in C. P. Zaleski and L. Vautrey, Le Reacteur Rapide Surregenerateur, Vols. 1 and 2, French Report CEA-2193, Commissariat à l'Énergie Atomique, 1961.)

The first leak in a radioactive circuit was in June 1966. The leak occurred in a drainage pipe where a crack was caused by restricted freedom of the pipe.

Sodium Heaters. The electric heating systems have burned out twice. In the first burnout, an electric heater burned through the gas tube connecting the gas cavities of the central tube and the reactor level-indicator tube. The reactor was shut down for repairs. When the reactor was started back up, coolant flow gradually diminished. All fuel-element assemblies had to be removed from the core and cleaned completely of oxide deposits before normal flow could be restored.

The second burnout occurred during a primary-circuit pumpdown. One of the drainage-section heaters burned out and caused the loss of both the heater and the vacuum in the system. The old heater was replaced with an insulating transformer. Rapid-disconnect devices were installed throughout to separate defective parts within 15 sec. No burnouts have occurred in the subsequent 8 years of operation.

Heat Exchangers. The two primary IHX's have logged about 30,000 hr of operation with no problems and no signs of leakage.

Steam Generators. An experimental once-through water–mercury–sodium steam generator was installed in June 1959. After 5 months of full-power operation, leaks appeared in the steam tubes of the evaporator section; thermal cycling and variable stresses in the downcomer tubes of the evaporator had induced intercrystalline corrosion. The Russians concluded that efforts to produce a completely safe steam generator *can* compromise reliability. They replaced the experimental steam generator with an air steam generator and have encountered no difficulties since. Some corrosion was observed at tube welds in 1966. Since it was so slight, the decision was made to observe the leaks rather than repair them. The areas seemed to leak under transients but were "self-sealing" under steady-state conditions.

Reactor Downtime. During the period 1959–1966, unscheduled shutdowns accounted for a total of 10,960 hr of reactor downtime, the largest amount of which was caused by the primary system and its components.[36]

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BONUS To Be Dismantled

As requested by the Puerto Rico Water Resources Authority (PRWRA) last May, the AEC ordered dismantling of the Boiling Water Nuclear Superheater (BONUS) reactor on Aug. 11, 1969.

The procedure to be used is included in the Safety Evaluation by the AEC Division of Reactor Licensing (Docket 115-4) which is reprinted below:

By application dated May 26, 1969, the Puerto Rico Water Resources Authority (PRWRA) requested the issuance of an order authorizing decommissioning of the BONUS reactor. The PRWRA's application is supported by a two-volume decommissioning plan. Volume I describes the plan for dismantling the facility and the final condition of the dismantled facility. Volume II is a safety analysis of the decommissioned plant. A previous amendment to the BONUS Operating Authorization No. DPRA-4 issued on Nov. 9, 1968, authorized PRWRA to possess but not to operate the BONUS reactor, and to perform certain work in preparation for dismantling the facility. The PRWRA plans to use the dismantled nuclear power plant for a period of 5 years following decommissioning as a public exhibition center.

Discussion

All new and irradiated fuel, control and shim rods, and all radioactive sources will be removed from the plant site except 18 depleted antimony pellets containing 8.5 curies of ^{60}Co which will be entombed in the plant. All radioactive waste will be removed from the site.

The reactor pressure vessel, neutron shield tank, pressure-vessel internals, primary coolant recirculation pumps, and any highly contaminated components and piping will be entombed within the biological shield. All penetrations through the biological shield will be severed, closed, and sealed.

Two areas within the biological shield, the pressure-vessel cavity and the recirculation pump room beneath the pressure vessel, will be used as entombment areas. Most of the biological shield is housed in a ¼-in. steel-plate casing. The casing terminates circumferentially where it intersects the fuel storage pool and the pipe chase wall. A projecting extension on the upper north face of the biological shield, previously used as a pipe chase, will be filled with concrete. The weight of the concrete-filled pipe chase will be transmitted to the foundation by added support columns. The entombment will be closed at the top by a 2½-ft-thick reinforced-concrete plug. A 2-in.-thick steel plate will be installed above the top concrete plug as a shock absorber with about an 8-in. air gap between the steel plate and concrete plug.

The reactor vessel will be sealed by bolting on a new flat closure head utilizing the existing head bolts and nuts and two silver-plated O-rings. The reactor vessel will be pressure tested, dried, purged, and filled with argon gas at atmospheric pressure.

The steel access door to the pump room from the feedwater heater room will be welded closed, and the opening in the wall on the pump-room side of the door will be filled with concrete. The removable 4- by 7-ft rectangular portion of the pump-room wall will be replaced by a 3.5-ft-thick concrete section, and a ¼-in.-thick steel cover plate will be welded to the entombment system steel outer liner.

The 18-in.-wide annular moat surrounding the entombment system and extending from elevation 7½ ft to elevation 16 ft will be filled with concrete. Critical joints such as the basement floor to entombment steel-liner joint will be sealed to prevent inleakage of water.

All piping, conduit, ducts, tubing, instrument lines, and sleeves penetrating the outer surface of the entombment system will be closed and sealed to prevent access into the penetrations or into the interior of the entombment area. All closures will be made without reducing the structural integrity of the entombment system.

Slightly contaminated systems, external to the entombment boundary, will be decontaminated and will remain in place. A final radiation survey of the entire plant, external to the entombment structure, will be performed at the conclusion of the decommissioning program to verify that radiation levels are acceptable.

Evaluation

Radiological Analysis of the Entombment System

The minimum design life of 140 years for the entombment structure is based on the potential contact and ingestion doses from the entombed radioactive materials.

Contact Dose. The BONUS entombment structure will initially contain approximately 50,000 curies of radioactive materials, consisting almost entirely of activation products. The dose rate at the entombment surface will not exceed 0.4 mr/hr at 1 cm except that "hot spots" up to 1.0 mr/hr will be permitted so long as the average radiation level of the surface does not exceed 0.2 mr/hr.

The most radioactive components contained within the entombment structure will have decayed to such an extent after 140 years that their direct dose rate would be about 0.2 mr/hr. We have concluded that the proposed design of the entombment structure is adequate to last at least 140 years under normal conditions and that the radioactivity will have decayed to acceptable levels by that time.

Ingestion Dose. In the ingestion-dose analysis performed by the licensee, it was conservatively assumed that the most active nuclide of the entombed components would be dissolved in water immediately at the end of the entombment life. The PRWRA has defined the design life requirement of the entombment structure as the period of time required for the limiting nuclide to decay to a level at which a solution saturated with this dissolved element would not exceed the maximum permissible concentration (MPC) or safe specific activity (SSA) if ingested. The SSA is the specific activity of an element which, upon replacement of all of that element in a person's body with the activated element, would result in a body burden no greater than the maximum permissible body burden (MPBB).

The most active nuclide in the entombed components was found to be the ^{63}Ni in the superheater. The SSA for ^{63}Ni showed that the entombment structure should have a lifetime of 126 years vs. the 140 years selected for the contact dose.

Although the SSA concept does not account for the effects of a number of variables that should be considered in other applications, we agree with the licensee that it provides a valid basis for evaluating the potential radioactive ingestion hazards in the deactivated BONUS facility. We also agree with PRWRA's ingestion dose analysis and consider the assumptions used in calculating the design life of the entombment to be conservative.

Design Basis Accident. The design basis accident (DBA) postulated by PRWRA for the decommissioned BONUS plant would result from a severe earthquake at the site followed by a tidal wave which floods the site. The earthquake was assumed to crack the containment building, the steel entombment liner, the concrete wall around the pressure vessel, the shield tank, the grout between the shield tank and the pressure vessel, and, finally, the bottom of the pressure vessel. The containment building foundation was assumed to remain watertight and to be flooded to the level of the highest adjacent ground, which is about 2 ft above the bottom of the pressure vessel. It was also assumed that the pressure-vessel internals had been corroding at a conservatively high rate and the corrosion products would be released instantaneously into the flood water when the building is flooded.

Each isotope has a unique time of maximum release which depends on the assumed corrosion rate within the entombment and the radioactive half-life for that particular isotope. The PRWRA's analysis shows that the only isotope which exceeds its occupational MPC at the time of its maximum release is ^{63}Ni. The bone limiting MPC for ^{63}Ni is 3×10^{-4} $\mu c/cm^3$. As a result of the DBA, the maximum concentration of ^{63}Ni would be 8.49×10^{-4} $\mu c/cm^3$. However, using the SSA concept, the maximum amount of ^{63}Ni which can be released with 1 g of nickel is 357 μc, and the standard man contains less than 0.01 g of nickel. Therefore, if all the nickel in a man were replaced by the Ni–^{63}Ni mixture present in the flood water, he would retain 3.57 μc of ^{63}Ni. This is well below the MPBB of 200 μc for ^{63}Ni. The dose to the gastro-

intestinal tract, which is not considered in the SSA, will not be excessive since the concentration does not exceed the limit of 0.01 $\mu c/cm^3$ for this dose. We agree with PRWRA's dose analysis, and we consider the potential ingestion doses resulting from the DBA to be acceptable.

Radiological Analysis of the External Systems

Contact Dose. Excessively contaminated portions of systems external to the entombment will be removed and entombed or shipped off-site for burial.

Remaining portions of external systems will be decontaminated to meet the criteria described previously. A postdecontamination radiation survey of all external systems will be performed to assure that those criteria are met. We have concluded that, if the criteria described earlier are met, the radiation levels will be acceptable for use of the facility as a public exhibition center.

Ingestion Dose. An accident involving the release of radioactivity from failure of a pipe external to the entombment was analyzed by PRWRA. We agree with its conclusion that the contamination levels of the external systems are sufficiently low that it is incredible that enough material could be released and ingested to exceed the MPBB.

Conclusion

On the basis of our review of the proposed BONUS decommissioning plan, we have concluded that there is reasonable assurance that dismantling of the facility and entombment or disposal of its highly radioactive components as proposed will not endanger the health and safety of the public. We recommend that PRWRA submit a postdecommissioning surveillance program with its final report on the status of the dismantled plant.

Donald J. Skovholt
Assistant Director for Reactor Operations
Division of Reactor Licensing

Date: Aug. 11, 1969

Nuclear Science Abstracts

The U. S. Atomic Energy Commission, Division of Technical Information, publishes *Nuclear Science Abstracts (NSA)*, a semimonthly journal containing abstracts of the literature of nuclear science and engineering.

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